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# Appendix H

Potential Repository Accident  
Scenarios: Analytical Methods  
and Results

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## **APPENDIX H. POTENTIAL REPOSITORY ACCIDENT SCENARIOS: ANALYTICAL METHODS AND RESULTS**

This appendix describes the methods and detailed results of the analysis the U.S. Department of Energy (DOE) performed for the Yucca Mountain Repository Environmental Impact Statement (EIS) to assess radiological impacts from potential accident scenarios at the proposed repository. The methods apply to repository accidents that could occur during preclosure only, including operation and monitoring, retrieval, and closure. In addition, this appendix describes the details of calculations for specific accidents that the analysis determined to be credible. Appendix J describes the analytical methods and results for accidents that could occur at the 72 commercial and 5 DOE sites and during transportation to the proposed repository.

The accident scenarios in this analysis, and the estimated impacts, are based on current information from the repository design (TRW 1999a, all). The results are based on assumptions and analyses that were selected to ensure that the impacts from accident scenarios are not likely to be underestimated. DOE has not developed the final design and operational details for the repository, and these details could result in lower impacts. The Department is currently engaged in preliminary efforts to identify accidents and evaluate their impacts as required to support the License Application for the repository that it will send to the Nuclear Regulatory Commission, and to show that the repository would comply with appropriate limits on radiation exposure to workers and the public from accidental releases of radionuclides. The final design could include additional systems and operational requirements to reduce the probability of accidents and to mitigate the release of radionuclides to ensure compliance with these safety requirements. The results from the accident analysis to meet licensing requirements would be more specific and comprehensive than those discussed in this appendix and would reflect final repository design and operational details.

### **H.1 General Methodology**

Because of the large amount of radioactive material to be handled at the proposed repository (see Appendix A), the focus of the analysis was on accident scenarios that could cause the release of radioactive material to the environment. The methodology employed to estimate the impact of accidents involving radioactive material included (1) evaluation of previous accident analyses performed for the repository, (2) identification of bounding accidents (reasonably foreseeable accidents with the maximum consequences) from the previous analyses, (3) identification of other credible accidents the previous analyses did not evaluate, (4) analyses of the selected accidents to determine the amount of radioactive material an accident could release to the environment, and (5) estimation of the consequences of the release of radioactive material in terms of health effects to workers and the public.

The analysis approach involved identifying bounding accidents (that is, accidents with maximum consequences) for each operational phase of the proposed repository. The analysis evaluated the impacts for these accidents, assuming the accident occurred without regard to the estimated probability. Thus, the analysis provides the impacts that could occur for the worst credible accidents. The results do not represent risk estimates because the impacts do not include a consideration of accident probability, which in most cases is very low. The risk from all repository accidents would be likely to be far less than the low risk, which DOE estimated by assuming that all of the bounding (maximum consequence) accidents would occur.

Accident frequency estimates were derived to establish the credibility of accident sequences and were not used to establish risk. Estimates of accident frequency are very uncertain due to the preliminary nature of the currently available repository design information and would be more fully evaluated in the safety

analysis required to support a License Application for the repository. Based on the available design information, the accident analysis approach was used to ensure that impacts from accidents are not likely to be underestimated (whether they are low-probability with high-consequence accidents or high-probability with low-consequence accidents).

For accidents not involving radioactive materials, the analysis determined that application of accident statistics from other DOE operations provided a reasonable estimate of nonradiological accident impacts (see Section H.2.2).

## **H.2 Potential Repository Accident Scenarios**

The proposed Yucca Mountain Repository has been the subject of intense evaluations for a number of years. Some of these evaluations included in-depth considerations of preclosure accidents that could occur during repository operations. The EIS used these previous evaluations, to the extent they are applicable and valid, to aid in the identification of initiating events, develop sequences, and estimate consequences. The EIS groups accidents as radiological accidents (Section H.2.1) that involve the unplanned release of radioactive material, and nonradiological accidents that involve toxic and hazardous materials (Section H.2.2).

### **H.2.1 RADIOLOGICAL ACCIDENT SCENARIOS**

Previous analyses that considered impacts of radiological accidents during preclosure included evaluations by Sandia National Laboratories and others (Jackson et al. 1984, all; SNL 1987, all; Ma et al. 1992, all; BMI 1984, all), and include more recent evaluations (DOE 1996a,b, all; DOE 1997a,b all; Kappes 1998, all; TRW 1997a, all). These evaluations were reviewed to assist in this assessment of radiological impacts from accidents during repository operations. In addition, EISs that included accident evaluations involving spent nuclear fuel and high-level radioactive waste were reviewed and used as applicable (USN 1996, all; DOE 1995, all).

Radiological accidents involve an initiating event that can lead to a release of radioactive material to the environment. The analysis considered accidents separately for two types of initiating events: (1) internal initiating events that would originate in the repository and involve equipment failures or human errors, or a combination of both, and (2) external initiating events that would originate outside the facility and affect the ability of the facility to maintain confinement of radioactive or hazardous material. The analysis examined a spectrum of accidents, from high-probability/low-consequence accidents to low-probability/higher-consequence accidents.

#### **H.2.1.1 Internal Events – Waste Handling Building**

The most recent and comprehensive repository accident scenario analysis for internal events in the Waste Handling Building is presented in Kappes (1998, all). This analysis considered the other important applicable accidents that previous analyses identified. It performed an in-depth evaluation of all operations planned for the repository and identified bounding accidents (those with the highest estimated risk) for each operation. More than 150 accidents were selected for analysis in eight operational categories. The accidents were identified based on multiple sources, including the *Preliminary MGDS Hazards Analysis* (DOE 1996b, all), current facility design drawings, and discussions with design personnel. These 150 accidents were reduced to 16 bounding accidents by retaining accidents that would produce the highest doses for groups of similar events (Kappes 1998, page 35). DOE used event trees and fault tree evaluation to estimate frequencies for the accidents. A review of these evaluations indicated that they were valid for use in the EIS with a few exceptions (noted below).

The evaluation used to identify internal accidents did not evaluate criticality events quantitatively (Kappes 1998, page 34). Continuing evaluations are under way to assess the probability and consequences of a criticality event. The risk from criticality events, however, would be unlikely to exceed the risk from the bounding events considered below. This preliminary conclusion is based on several factors:

- The probability of a criticality event would be very low. This is based on the Nuclear Regulatory Commission design requirement (10 CFR Part 60) that specifies that two independent low-probability events must occur for criticality to be possible and that this requirement will be part of the licensing basis for the repository. On the basis of this requirement, the event is unlikely to be credible (Jackson et al. 1984, page 18). Further, a criticality event would require the assembly of fuel with sufficient fissionable material to sustain a criticality. Since the commercial spent-nuclear fuel to be handled at the repository is spent (that is, it has been used to produce power), the remaining fissionable material is limited. For the pressurized-water reactor fuel, the amount of fuel that contains sufficient fissionable material to achieve criticality is only a small percent spent nuclear fuel (DOE 1998a, page C-46). This material would have to be assembled in sufficient quantity to achieve criticality, and the moderator (water) would somehow have to be added to the assembled material. A quantitative estimate of criticality frequency is planned as part of the license application (Kappes 1998, page 34).
- The criticality event that could occur despite the preventive measures described above would be unlikely to compromise the confinement function of the ventilation and filtration system of the Waste Handling Building. These features would inhibit the release of particulate radionuclides. By contrast, the seismic event scenario (discussed in Section H.2.1.3) assumes failure of these mitigating features.
- Criticality could occur only if the material was moderated with water and had sufficient fissionable material in a configuration that could allow criticality. The water surrounding the material would act to inhibit the release of particulate material (DOE 1994, Volume 1, Appendix D, page F-85) and, thus, would limit the source term.
- During the monitoring and closure phase of operations, water needs to enter a waste package that contains fuel with sufficient fissionable material to go critical. Water would have to flood a drift and leak into a defective waste package to cause a criticality. Such an event is considered not credible due to the lack of sufficient water sources, detection and remediation of water in-leakage, and high-quality leak proof waste packages.

Considering these factors, the criticality event does not appear to be a large potential contributor to risk.

### **RISK**

Risk is defined as the possibility of suffering harm. It considers both the frequency (or probability) and consequences of an accident. In the scientific community, risk is usually defined and computed as the product of the frequency of an accident and the consequences that result. This is the definition of risk used in this analysis.

Rather than develop a single, overall expression of the risks associated with proposed actions, DOE usually finds it more informative in its EIS accident scenario analyses to consider a spectrum of accidents from low-probability, relatively high-consequence accidents to high-probability, low-consequence accidents. Nevertheless, risk is a valuable concept to apply in evaluating the spectrum of accident scenarios to ensure that accidents that are expected to dominate risk have been adequately considered.

Table H-1 lists the bounding accident scenarios identified in Kappes (1998, page 40). For each accident scenario, the table lists (1) the location of the accident, (2) the material at risk, or the amount of radioactive material involved in the accident, and (3) if the analysis assumed that filtration (high-efficiency particulate air filters) would be available to mitigate radioactive material releases. Filtration would be provided in most areas of the Waste Handling Building (TRW 1999b, page 41) and in the subsurface emplacement facilities (TRW 1999a, page 4-61). The Frequency column in Table H-1 lists the estimated annual frequency of the event (Kappes 1998, all). The last column indicates if the EIS analysis retained, eliminated, or adjusted details of the accident scenario.

**Table H-1.** Bounding internal accident scenarios for the Waste Handling Building and emplacement operations.

Location <sup>a</sup>	Number	Accident <sup>b</sup>	Material at risk <sup>c</sup>	Filters	Frequency	Disposition
A	1	6.9-meter drop of shipping cask	61 BWR assemblies	No	$4.5 \times 10^{-4}$	Retained
A	2	6.9-meter drop of shipping cask	61 BWR assemblies	Yes	-- <sup>d</sup>	Eliminated
A	3	7.1-meter drop of shipping cask	26 PWR assemblies	No	$6.1 \times 10^{-4}$	Retained
A	4	7.1-meter drop of shipping cask	26 PWR assemblies	Yes	--	Eliminated
A	5	4.1-meter drop of shipping cask	61 BWR assemblies	No	$1.4 \times 10^{-3}$	Retained
A	6	4.1-meter drop of shipping cask	61 BWR assemblies	Yes	--	Eliminated
A	7	4.1-meter drop of shipping cask	26 PWR assemblies	No	$1.9 \times 10^{-3}$	Retained
B	8	8.6-meter drop of canister	DOE high-level waste	Yes	$4.2 \times 10^{-5}$	Eliminated <sup>e</sup>
B	9	6.3-meter drop of multicannister overpack	N-Reactor fuel	Yes	$4.5 \times 10^{-4}$	Retained
B	10	6.3-meter drop of multicannister overpack	N-Reactor fuel	No	$2.2 \times 10^{-7}$	Added <sup>f</sup>
C	11	5-meter drop of transfer basket	8 PWR assemblies	Yes	$1.1 \times 10^{-2}$	Retained
C	12	5-meter drop of transfer basket	8 PWR assemblies	No	$2.8 \times 10^{-7}$	Added <sup>f</sup>
C	13	7.6-meter drop of transfer basket	16 BWR assemblies	Yes	$7.4 \times 10^{-3}$	Retained
C	14	7.6-meter drop of transfer basket	16 BWR assemblies	No	$1.9 \times 10^{-7}$	Added <sup>f</sup>
D	15	6-meter vertical drop of disposal container	21 PWR assemblies	Yes	$1.8 \times 10^{-3}$	Retained
D	16	6-meter vertical drop of disposal container	21 PWR assemblies	No	$8.6 \times 10^{-7}$	Added <sup>g</sup>
D	17	2.5-meter horizontal drop of disposal container	21 PWR assemblies	Yes	$3.2 \times 10^{-4}$	Eliminated <sup>g</sup>
E	18	Rockfall on waste package	44 BWR assemblies	No	$4.2 \times 10^{-8}$	Eliminated <sup>h</sup>
E	19	Transporter runaway and derailment	21 PWR assemblies	Yes	$1.2 \times 10^{-7}$	Retained <sup>i</sup>

a. Location designators: A = Cask/Carrier Transport and Handling Area, B = Canister Transfer System, C = Assembly Transfer System, D = Disposal Container Handling System, E = Waste Emplacement and Subsurface Facility.

b. To convert meters to feet, multiply by 3.2808.

c. BWR = boiling-water reactor; PWR = pressurized-water reactor.

d. Eliminated from evaluation because current design does not include a filter system for this area (Kappes 1998, page 40).

e. Eliminated on the basis that it would not be a risk contributor because the N-Reactor multicannister overpack drop (accident scenario B10) has an estimated frequency more than 10 times higher, and the N-Reactor fuel has a higher radionuclide inventory (Appendix A).

f. These accident scenarios, involving loss of filtration, were added because they would exceed the level of credibility recommended by DOE (frequency greater than  $1 \times 10^{-7}$  per year) (DOE 1993, page 28). The corresponding U.S. Nuclear Regulatory Commission limit (used in Kappes 1998, page 4) is  $1 \times 10^{-6}$  per year. The Commission considers accidents with frequencies less than  $1 \times 10^{-6}$  per year to be beyond design basis events.

g. Eliminated because it would not contribute to risk in comparison to accident scenario 15 at location D., a higher drop event that would produce larger consequences with a higher frequency.

h. Eliminated on the basis of low frequency, below the credible level of  $1 \times 10^{-7}$ .

i. Frequency adjusted to account for the filtration system in the current design.

The following paragraphs contain details of the postulated accident scenarios in each location.

#### **H.2.1.1.1 Cask/Carrier Transport and Handling Area**

These accidents (Table H-1, location A, accidents 1 through 7) would involve mishaps that could occur during the process of handling the transportation casks at the repository. The transportation casks would be designed to withstand impacts from collisions and drops, and this capability is augmented by impact limiters, which would be required during transportation. After cask arrival at the repository, the limiters would be removed to facilitate handling of the casks. The casks would then become more vulnerable to damage from physical impact. The analysis assumed that damage to the casks would occur if they were dropped from heights greater than the design basis of 2 meters (6.6 feet) (Kappes 1998, page 13) without the impact limiters. The various heights of the drops in the “Accident” column in Table H-1 correspond to the maximum height to which the casks could be lifted during the various operations the analysis assumed crane failure would occur. The material-at-risk column lists the contents of the casks when the accident occurred. The largest casks are designed to hold either 61 boiling-water reactor or 26 pressurized-water reactor fuel assemblies.

Accident scenarios from Kappes (1998) that assume a filtration system is available (accidents A2, A4, and A6) were eliminated from consideration in the EIS because the current design concept of the Cask/Carrier Transport and Handling Area does not include such a filtration system; they were considered in Kappes (1998, page 40) for information only.

#### **H.2.1.1.2 Canister Transfer System**

The Canister Transfer System would handle canisters that arrived at the repository and were suitable for direct transfer to the disposal container. The bounding accident scenarios for these operations would be canister drops of DOE high-level radioactive waste and N-Reactor fuel (accidents 8 and 9 at location B in Table H-1). The analysis eliminated the DOE high-level radioactive waste canister drop because it would not be a risk contributor in comparison to the N-Reactor fuel drop. The N-Reactor multiccanister overpack drop would have a frequency more than 10 times greater than that for the high-level radioactive waste canister drop, and the N-Reactor radionuclide inventory would be greater (see Appendix A). The EIS analysis added an additional accident scenario, which would be a drop of the N-Reactor fuel canister with loss of the filtration system. The analysis estimated the filtration system failure probabilities by using the fault tree analysis technique, and the results differ somewhat from the failures identified in Section H.2.1.1.3 due to design variations dependant on location in the surface facilities of the repository. DOE computed this accident scenario probability by combining the accident drop probability of 0.00045 with the filter system failure of  $4.8 \times 10^{-4}$  from Kappes (1998, page 4) for an accident sequence frequency of  $2.2 \times 10^{-7}$  per year. [Kappes (1998, page 4) did not consider accident sequences with frequencies less than  $1 \times 10^{-6}$ .] This sequence frequency is based on failure of the heating, ventilating, and air conditioning system such that it would not provide filtration for 24 hours following the accident, consistent with Kappes (1998, page VIII-1).

#### **H.2.1.1.3 Assembly Transfer System**

The Assembly Transfer System would handle bare, intact commercial spent nuclear fuel assemblies from pressurized- and boiling-water reactors. The assemblies would be unloaded from the transportation cask in the cask unloading pool. Next, they would be moved to the assembly staging pool where they would be placed in baskets that contained either four pressurized-water reactor assemblies or eight boiling-water assemblies. The baskets would be moved from the pool and transferred to the assembly drying station from which they would be loaded, after drying, in the disposal containers. The bounding accident scenarios found during a review of this operation (Kappes 1998, page 40) were drops of a suspended basket loaded with fuel assemblies on another loaded basket in the drying vessel (accident scenarios 11 and 13 at location C from Table H-1). DOE added two accident scenarios to the EIS analysis that

included failure of the high-efficiency particulate air filtration system (accident scenarios 12 and 14 at location C from Table H-1). DOE computed the frequency of these accidents by combining the accident drop frequency with the filter failure probability of 0.000025, which corresponds to the failure probability of the heating, ventilation, and air conditioning system in the assembly transfer area (Kappes 1998, page 11). Thus, the frequency of a drop accident and subsequent failure of the heating, ventilation, and air conditioning system during the 24 hours (the period assumed that the filtration system would need to operate to remove the particulate material effectively) would be:

- For boiling-water reactor assembly drop:  $0.011 \times 0.000025 = 0.00000028$
- For pressurized-water reactor assembly drop:  $0.0074 \times 0.000025 = 0.00000019$

#### **H.2.1.1.4 Disposal Container Handling System**

The Disposal Container Handling System would prepare empty disposal containers for the loading of nuclear materials, transfer disposal containers to and from the assembly and canister transfer systems, weld the inner and outer lids of the disposal containers, and load disposal containers on the waste emplacement transporter. After the disposal container had been loaded and sealed, it would become a waste package. Disposal containers would be lifted and moved several times during the process of preparing them for loading on the waste emplacement transporter. DOE examined the details of these operations and identified numerous accident scenarios that could occur (Kappes 1998, Attachment V). The bounding accident scenarios from this examination would be the disposal container drop accident scenarios listed as accident scenarios 15 and 17 at Location D in Table H-1. However, the analysis eliminated accident scenario 17 because it would be a minor contributor to risk in comparison to accident scenario 15. Accident scenario 15, which would have a higher probability (by about a factor of 6), would produce a higher radionuclide release due to the increased drop height (by a factor of more than 2). Thus, the overall risk contribution from accident scenario 17 would be less than 10 percent of the risk from accident scenario 15. For the EIS, DOE added another accident scenario (16) to account for the possibility of loss of filtration. The analysis assumed that the heating, ventilation, and air conditioning filtration system would fail with a probability of 0.00048 (Kappes 1998, page 4).

#### **H.2.1.1.5 Waste Emplacement and Subsurface Facility Systems**

The waste emplacement system would transport the loaded and sealed waste package from the Waste Handling Building to the subsurface emplacement area. This system would operate on the surface between the North Portal and the Waste Handling Building, and in the underground ramps, main drifts (tunnels), and emplacement drifts. It would use a reusable railcar for waste package transportation. The railcar would be moved into the waste emplacement area by an electric locomotive, and the waste package would be placed in the emplacement drift. The bounding accident scenarios identified (Kappes 1998, page 40) for this operation would be accident scenarios 18 and 19 at location E, as listed in Table H-1. However, DOE eliminated accident scenario 18 (rockfall on waste package) because the estimated frequency of a radioactive release from such an event is not credible (estimated frequency of  $4.2 \times 10^{-8}$  per year) (Kappes 1998, page VI-5).

An accident scenario involving a failure of the ventilation system in conjunction with a transporter runaway and collision (accident scenario F19 from Table H-1) would not be credible, so the sequence was not analyzed. The original transporter runaway and derailment accident scenario assumed the involvement of 44 boiling-water reactor assemblies (Kappes 1998, page 40). The EIS analysis assumed the involvement of 21 pressurized-water reactor assemblies because (1) they would represent a slightly higher impact potential due to the greater radionuclide inventory than that in the smaller 44 boiling-water reactor assemblies and would, therefore, bound the equivalent accident involving such assemblies, and

(2) an accident scenario involving pressurized-water reactor fuel would be more likely because DOE expects to emplace about twice as much of this type of fuel in the proposed repository (Appendix A).

Section H.2.1.4 describes the source terms (amount and type of radionuclide release) for these accident scenarios, and Section H.2.1.5 assesses the estimated consequences from the accident scenarios.

### **H.2.1.2 Internal Events – Waste Treatment Building**

An additional source of radionuclides could be involved in accidents in the Waste Treatment Building. This building, which would be connected to the northeast end of the Waste Handling Building, would house the Site-Generated Radiological Waste Handling System (TRW 1999b, page 37). This system would collect site-generated low-level radioactive solid and liquid wastes and prepare them for disposal. The radioactivity of the waste streams would be low enough that no special features would be required to meet Nuclear Regulatory Commission radiation safety requirements (shielding and criticality) (TRW 1999b, page 38).

The liquid waste stream to the Waste Treatment Building would consist of aqueous solutions that could contain radionuclides resulting from decontamination and washdown activities in the Waste Handling Building. The liquid waste would be evaporated, mixed with cement (grouted), and placed in 0.21-cubic-meter (55-gallon) drums for shipment off the site (TRW 1999b, page 53). The evaporation process would reduce the volume of the liquid waste stream by 90 percent (DOE 1997c, Summary).

The solid waste would consist of noncompactible and compactible materials and spent ion-exchange resins. These materials ultimately would be encapsulated in concrete in 0.21-cubic meter (55-gallon) drums after appropriate processing (TRW 1999b, page 55).

Water in the Assembly Staging Pools of the Waste Handling Building would pass through ion exchange columns to remove radionuclides and other contaminants. These columns would accumulate radionuclides on the resin in the columns. When the resin is spent (unable to remove radionuclides effectively from the water), the water flow would be diverted to another set of columns, and the spent resin would be removed and dewatered for disposal as low-level waste or low-level mixed waste. These columns could have external radiation dose rates associated with them because of the activation and fission product radionuclides accumulated on the resins. They would be handled remotely or semiremotely. During the removal of the resin and preparation for offsite shipment in the Waste Treatment Building, an accident scenario involving a resin spill could occur. However, because the radionuclides would have been chemically bound to the resin in the column, an airborne radionuclide release would be unlikely. Containment and filter systems in the Waste Treatment Building would prevent exposure to the public or noninvolved workers. Some slight exposure of involved workers could occur during the event or during recovery operations afterward. DOE made no further analysis of this event.

Because there is no detailed design of the Waste Treatment Building at present and operational details are not yet available, DOE used the recent Waste Management Programmatic EIS (DOE 1997c, all) and supporting documentation (Mueller et al. 1996, all) to aid in identifying potential accident scenarios and evaluating radionuclide source terms. For radiological impacts, the analysis focused on accident scenarios with the potential for airborne releases to the atmosphere. The liquid stream can be eliminated because it has a very low potential for airborne release; the radionuclides would be dissolved and energy sources would not be available to disperse large amounts of the liquid into droplets small enough to remain airborne. Many low-level waste treatment operations, including evaporation, solidifying (grouting), packaging, and compaction can be excluded because they would lack sufficient mechanistic stresses and energies to create large airborne releases, and nuclear criticalities would not be credible for

low-level waste (Mueller et al. 1996, page 13). Drum-handling accidents are expected to dominate the risk of exposure to workers (Mueller et al. 1996, page 93).

The estimated frequency of an accident involving drum failure is about 0.0001 failure per drum operation (Mueller et al. 1996, page 39). The total number of drums containing grouted aqueous waste would be 2,280 per year (DOE 1997c, page 30). The analysis assumed that each drum would be handled twice, once from the Waste Treatment Building to the loading area, and once to load the drum for offsite transportation. Therefore, the frequency of a drum failure involving grouted aqueous waste would be:

$$\begin{aligned}\text{Frequency} &= 2,280 \text{ aqueous (grouted) low-level waste drums per year} \\ &\quad \times 2 \text{ handling operations per drum} \\ &\quad \times 0.0001 \text{ failure per handling operation} \\ &= 0.46 \text{ aqueous (grouted) low-level waste drum failures per year.}\end{aligned}$$

The number of solid-waste grouted drums produced would be 2,930 per year (DOE 1997c, page 35). Assuming two handling operations and the same failure rate yields a frequency of drum failure of:

$$\begin{aligned}\text{Frequency} &= 2,930 \text{ solid low-level waste drums per year} \\ &\quad \times 2 \text{ handling operations per drum} \\ &\quad \times 0.0001 \text{ failure per handling operation} \\ &= 0.59 \text{ solid low-level waste drum failures per year.}\end{aligned}$$

Failure of these drums would result in a release of radioactive material, which later sections evaluate further.

### H.2.1.3 External Events

External events are either external to the repository (earthquakes, high winds, etc.) or are natural processes that occur over a long period of time (corrosion, erosion, etc.). DOE performed an evaluation to identify which of these events could initiate accidents at the repository with potential for release of radioactive material.

Because some external events evaluated as potential accident-initiating events would affect both the Waste Treatment and Waste Handling Buildings simultaneously [the buildings are physically connected (TRW 1999b, page 38)], this section considers potential accidents involving external event initiators, as appropriate, for the combined buildings.

Table H-2 lists generic external events developed as potential accident initiators for consideration at the proposed repository and indicates how each potential event could relate to repository operations based on an initial evaluation process. The list, from DOE (1996b, page 15), was developed by an extensive review of relevant sources and known or predicted geologic, seismologic, hydrologic, and other characteristics. The list includes external events from natural phenomena as well as man-caused events.

The center column in Table H-2 (relation to repository) represents the results of a preliminary evaluation to determine the applicability of the event to the repository operations, and is based in part on evaluations previously reported in DOE (1996b, all). Events were excluded for the following reasons:

- Not applicable because of site location (condition does not exist at the site)
- Not applicable because of site characteristics (potential initiator does not exist in the vicinity of the site)

**Table H-2.** External events evaluated as potential accident initiators.<sup>a</sup>

Event	Relation to repository <sup>b</sup>	Comment
Aircraft crash	A	
Avalanche	C	
Coastal erosion	B	
Dam failure	C	
Debris avalanche	A	Caused by excessive rainfall
Dissolution	A	Chemical weathering of rock
Epeirogenic displacement (tilting of the Earth's crust)	D (earthquake)	Large-scale surface uplifting and subsidence
Erosion	D (flooding)	
Extreme wind	A	
Extreme weather	A	Includes extreme episodes of fog, frost, hail, ice cover, etc.
Fire (range)	A	
Flooding	A	
Denudation	E	Wearing away of ground surface by weathering
Fungus, bacteria, algae	E	A potential waste package long-term corrosion process not relevant during the repository operational period <sup>c</sup>
Glacial erosion	B	
High lake level	C	
High tide	B	
High river stage	C	
Hurricane	B	
Inadvertent future intrusion	E	To be addressed in postclosure Performance Assessment
Industrial activity	A	
Intentional future intrusion	E	
Lightning	A	
Loss of offsite or onsite power	A	
Low lake level	C	
Meteorite impact	A	
Military activity	A	
Orogenic diastrophism	D (earthquake)	Movement of Earth's crust by tectonic processes
Pipeline rupture	C	
Rainstorm	D (flooding)	
Sandstorm	A	
Sedimentation	B	
Seiche	B	Surface water waves in lakes, bays, or harbors
Seismic activity, uplift	D (earthquake)	
Seismic activity, earthquake	A	
Seismic activity, surface fault	D (earthquake)	
Seismic activity, subsurface fault	D (earthquake)	
Static fracture	D (earthquake)	Rock breakup caused by stress
Stream erosion	B	
Subsidence	D (earthquake)	Sinking of Earth's surface
Tornado	A	
Tsunami	B	Sea wave caused by ocean floor disturbance
Undetected past intrusions	E	
Undetected geologic features	D (earthquake, volcanism ash fall)	
Undetected geologic processes	D (erosion, earthquake, volcanism ash fall)	
Volcanic eruption	D (volcanism ash fall)	
Volcanism, magmatic	D (volcanism ash fall)	
Volcanism, ash flow	D (volcanism ash fall)	
Volcanism, ash fall	A	
Waves (aquatic)	B	

a. Source: DOE (1996b, page 15).

b. A = retained for further evaluation; B = not applicable because of site location; C = not applicable because of site characteristics (threat of event does not exist in the vicinity of the site); D = included in another event as noted; E = does not represent an accident-initiating event for proposed repository operations.

c. Source: TRW (1999a, all).

- Included in another event
- Does not represent an accident-initiating event for proposed repository operations

The second column of Table H-2 identifies the events excluded for these reasons. The preliminary evaluation retained the events identified in Table H-2 with “A” for further detailed evaluation. The results of this evaluation are as follows:

- 1. Aircraft Crash.** The EIS analysis evaluated the frequency of aircraft crashes on the proposed repository to determine if such events could be credible and, therefore, candidates for consequence analysis. This frequency determination used analytical methods recommended for aircraft crashes into hazardous facilities (DOE 1996c, all).

An earlier analysis assumed that the only reasonable aircraft crash threat would be from military aircraft operations originating from Nellis Air Force Base (Kimura, Sanzo, and Sharirli 1998, page 8), primarily because commercial and general aviation aircraft are restricted from flying over the Nevada Test Site. DOE considered this assumption valid and adopted it for the EIS analysis.

The formula used in the crash frequency analysis, taken from Kimura, Sanzo, and Sharirli (1998, pages 9 to 12) based on DOE (1996c, all), was:

$$F = (N_t \div A_t) \times A_{\text{eff}} \times \lambda \times (4 \div \pi) \times (R_{\text{eff}} + R_c)$$

where:

- F = the frequency per year of aircraft crashes on the repository
- N<sub>t</sub> = total number of aircraft overflights per year
- A<sub>t</sub> = total area of the overflight region
- A<sub>eff</sub> = effective area of the repository (target area)
- λ = crash rate of the aircraft per mile of flight
- R<sub>eff</sub> = effective radius of the repository (target area)
- R<sub>c</sub> = radius of the crash area potentially affected by a distressed aircraft

The parameters in this formula were quantified as follows:

- N<sub>t</sub> The estimated total number of flights in the flight corridor in the vicinity of the repository would be 13,000 per year, with the repository located on the western edge of the corridor, which extends 49 kilometers (30 miles) to the east. Most flights would not be observed from the repository. However, this value was used in a recent crash assessment for a Nevada Test Site facility beneath the same airspace as the repository (Kimura, Sanzo, and Sharirli 1998, page 7). Future Nellis operations could result in increased overflights. The only known planned change in future activities involve the bed-down of F-22 fighter aircraft. This planned activity involves 17 aircraft that will be at Nellis by 2010. The additional aircraft would increase flight activities by only 2 to 3 percent over current activities (Myers 1997, page 2).
- A<sub>t</sub> The total area of the overflight area would be about 3,400 square kilometers (1,300 square miles) (Kimura, Sanzo, and Sharirli 1998, page 18).

$A_{\text{eff}}$  The analysis estimated the repository target area by assuming that the roof of the Waste Handling Building would be the only vulnerable location at the repository with the potential for a large radionuclide release as a result of an aircraft impact. This is because the Waste Handling Building would be the only facility that would handle bare spent nuclear fuel assemblies. The shipping casks and the waste packages loaded with spent nuclear fuel or high-level radioactive waste would not be vulnerable to air crash impacts because both would have steel walls thick enough to prevent aircraft penetration. The Waste Treatment Building would not contain large amounts of radioactive material, so radionuclide releases from accidents involving this building would not produce large impacts (see Section H.2.1.4 for details). Further, the walls of the Waste Handling Building around areas for the handling of canisters and fuel assemblies would be 1.5 meters (5 feet) thick to a level of 9 meters (30 feet), and then 1 meter (3.3 feet) thick to the intersection with the roof (TRW 1999b, pages 31 to 37). The aircraft crash would not penetrate these walls because the concrete penetration capability for aircraft is limited to about 0.76 meter (2.5 feet) (see Appendix K for details). Therefore, the only likely vulnerable target area at the repository would be the roof of the Waste Handling Building, which would consist of concrete 20 to 25 centimeters (8 to 10 inches thick) (TRW 1999b, pages 31 to 37). The overall footprint of the Waste Handling Building would be about 163 meters by 165 meters (535 feet by 540 feet), which would produce a target area of approximately 27,000 square meters (290,000 square feet).

$\lambda$  The crash rate for the small military aircraft involved in the overflights [primarily F-15s, F-16s, and A-10s (USAF 1999, pages 1-34 to 1-35)] would be  $1.14 \times 10^{-8}$  per kilometer ( $1.84 \times 10^{-8}$  per mile) (Kimura, Sanzo, and Sharirli 1998, page 7). Large military aircraft fly over the area to some extent, but have a lower crash rate [ $1.17 \times 10^{-9}$  per kilometer ( $1.9 \times 10^{-9}$  per mile) (Kimura, Sanzo, and Sharirli 1998, page 7)]. Thus, the use of the small aircraft crash rate bounds the large aircraft crash rate.

$R_{\text{eff}}$  The effective radius of the repository is the equivalent radius of the repository target effective area ( $A_{\text{eff}}$ ), or  $R_{\text{eff}}$  is equal to the square root of the quotient 27,000 square meters divided by  $\pi$ , which is about 93 meters (310 feet).

$R_c$  The radius of the crash area potentially affected by a distressed military aircraft represents the distance an aircraft could travel after engine failure (glide distance). This distance is the glide ratio of the aircraft times the elevation of the flight above the ground. The aircraft are required to fly a minimum of 4,300 meters (14,000 feet) above mean sea level while in the airspace over the repository (Kimura, Sanzo, and Sharirli 1998, page 5). The actual altitude flown varies from 4,600 to 7,000 meters (15,000 to 23,000 feet) (Tullman 1997, page 4). For this analysis, a mean altitude of 5,800 meters (19,000 feet) was assumed. Because the Waste Handling Building would be at about 1,100 meters (3,680 feet) (TRW 1998a, page I-6), the mean flight elevation for aircraft above the repository ground level would be about 4,700 meters (10,000 feet). The glide ratio for the aircraft involved in the overflights (F-15, F-16, and A-10) is 8 (Thompson 1998, all). Therefore,  $R_c$  would be 4,700 meters multiplied by 8, which is 38,000 meters or 38 kilometers (23 miles).

Substituting these values into the frequency equation yields:

$$\begin{aligned} F &= (13,000 \div 3,400) \times 0.027 \times 1.14 \times 10^{-8} \times (4 \div \pi) \times (38 + 0.093) \\ &= 5.6 \times 10^{-8} \text{ crash per year.} \end{aligned}$$

Thus, aircraft crashes on the vulnerable area of the repository are not credible because the probability would be below  $1 \times 10^{-7}$  per year, which is the credible limit specified by DOE (1993, page 28).

2. **Debris Avalanche.** This event, which can result from persistent rainfall, would involve the sudden and rapid movement of soil and rock down a steep slope. The nearest avalanche potential to the proposed location for the Waste Handling Building is Exile Hill (the location of the North Portal entrance). The base of Exile Hill is about 90 meters (300 feet) from the location of the Waste Handling Building. Since Exile Hill is only about 30 meters (100 feet) high (TRW 1997a, page 5.09), it would be unlikely that avalanche debris would reach the Waste Handling Building. Furthermore, the design for the Waste Handling Building includes concrete walls about 1.5 meters (5 feet) thick (TRW 1999b, page 38) that would provide considerable resistance to an impact or buildup of avalanche debris.
3. **Dissolution.** Chemical weathering could cause mineral and rock material to pass into solution. This process, called dissolution, has been identified as potentially applicable to Yucca Mountain (DOE 1996b, page 18). However, this is a very slow process, which would not represent an accident-initiating event during the preclosure period being considered in this appendix.
4. **Extreme Wind.** Extreme wind conditions could cause transporter derailment (TRW 1997b, page 72), the consequences of which would be bounded by a transporter runaway accident scenario. The runaway transporter accident scenario is discussed further in Section H.2.1.4.
5. **Extreme Weather.** This potential initiating event includes various weather-related phenomena including fog, frost, hail, drought, extreme temperatures, rapid thaws, ice cover, snow, etc. None of these events would have the potential to cause damage to the Waste Handling Building that would exceed the projected damage from the earthquake event discussed in this section. In addition, none of these events would compromise the integrity of waste packages exposed on the surface during transport operations. Thus, the earthquake event and other waste package damage accident scenarios considered in this appendix would bound all extreme weather events. It would also be expected that operations would be curtailed if extreme weather conditions were predicted.
6. **Fire.** There would be two potential external fire sources at the repository site—diesel fuel oil storage tank fires and range fires. Diesel fuel oil storage tanks would be some distance [more than 90 meters (300 feet)] from the Waste Handling Building and Waste Treatment Building (TRW 1999b, Attachment IV Figure 4). Therefore, a fire at those locations would be highly unlikely to result in any meaningful radiological consequences. Range fires could occur in the vicinity of the site, but would be unlikely to be important accident contributors due to the clearing of land around the repository facilities. Furthermore, the potential for early fire detection and, if necessary, active fire protection measures and curtailment of operations (TRW 1999b, Section 4.2) would minimize the potential for fire-initiated radiological accidents. DOE is performing detailed evaluations of fire-initiating events (Kappes 1998, page III-2), and will incorporate the results in the Final EIS as appropriate.
7. **Flooding.** Flash floods could occur in the vicinity of the repository (DOE 1996b, page 21). However, an earlier assessment (Kappes 1998, page 32) screened out severe weather events as potential accident-initiating events primarily by assuming that operational rules will preclude transport and emplacement operations whenever there are local forecasts of severe weather. A quantitative analysis of flood events (Jackson et al. 1984, page 34) concluded that the only radioactive material that extreme flooding would disperse to the environment would be decontamination sludge from the waste treatment complex. The doses resulting from such dispersion would be limited to workers, and would be very small (Jackson et al. 1984, page 53). A more recent study reached a similar conclusion (Ma et al. 1992, page 3-11).
8. **Industrial Activity.** This activity would involve both drift (tunnel) development activities at the repository and offsite activities that could impose hazards on the repository.

- a. **Emplacement Drift Development Activities** – Drift development would continue during waste package emplacement activities. However, physical barriers in the main drifts would isolate development activities from emplacement activities (TRW 1999a, page 4-52). Thus, events that could occur during drift development activities would be unlikely to affect the integrity of waste packages.
  - b. **External Industrial Activities** – The analysis examined anticipated activities in the vicinity of the proposed repository to determine if accident-initiating events could occur. Two such activities—the Kistler Aerospace activities and the Wahmonie rocket launch facility—could initiate accidents at the repository from rocket impacts. The Wahmonie activities, which involved rocket launches from a location several miles east of the repository site, have ended (Wade 1998, all), so this facility no longer poses a risk to the repository. The planned Kistler Aerospace activities would involve launching rockets from the Nevada Test Site to place satellites in orbit (DOE 1996d, Volume 1, page A-42). However, at present there is insufficient information to assess if this activity would pose a threat to the repository. As details become available, the Final EIS will evaluate the potential for this activity to become an external accident-initiating event. (Aircraft activity was discussed in item 1 above.)
9. **Lightning.** This event has been identified as a potential design-basis event (DOE 1997b, pages 86 and 87). Therefore, the analysis assumed that the designs of appropriate repository structures and transport vehicles would include protection against lightning strikes. The lightning strike of principal concern would be the strike of a transporter train during operations between the Waste Handling Building and the North Portal (DOE 1997b, page 86). The estimated frequency of such an event would be  $1.9 \times 10^{-7}$  per year (Kappes 1998, page 33). DOE expects to provide lightning protection for the transporter (TRW 1998b, Volume 1, page 18) such that a lightning strike that resulted in enough damage to cause a release would be well below the credibility level of  $1 \times 10^{-7}$  per year (DOE 1993, page 28).
10. **Loss of Offsite Power.** A preliminary evaluation (DOE 1997b, page 84) concluded that a radionuclide release from an accident sequence initiated by a loss of offsite power would be unlikely. Loss of offsite power events could result in a failure of the ventilation system and of the overhead crane system. However, there would be emergency power for safety systems at the site (TRW 1999b, page 45). Loss of offsite power was included as a contributor to the frequency of crane failure (Kappes 1998, page III-6), as listed in the event frequencies in Table H-1.
11. **Meteorite Impact.** This event would not be credible based on a strike frequency of  $2 \times 10^{-8}$  per year for a damaging meteorite [based on data in Solomon, Erdmann, and Okrent (1975, page 68)]. This estimate accounts for the actual area of the Waste Handling Building roof given previously in item 1.
12. **Military Activity.** Two different military activities would have the potential to affect repository operations. One is the possibility of an aircraft crash from overflights from Nellis Air Force Base. The analysis determined that this event would not be credible, as described above in this section. The second potential activity is the resumption of underground nuclear weapons testing, which the United States has suspended. The only impact such testing could impose on the repository would be ground motion associated with the energy released from the detonation of the weapon. The impact of such motion was the subject of a recent study that concluded that ground motions at Yucca Mountain from nuclear tests would not control seismic design criteria for the potential repository (Walck 1996, page i).

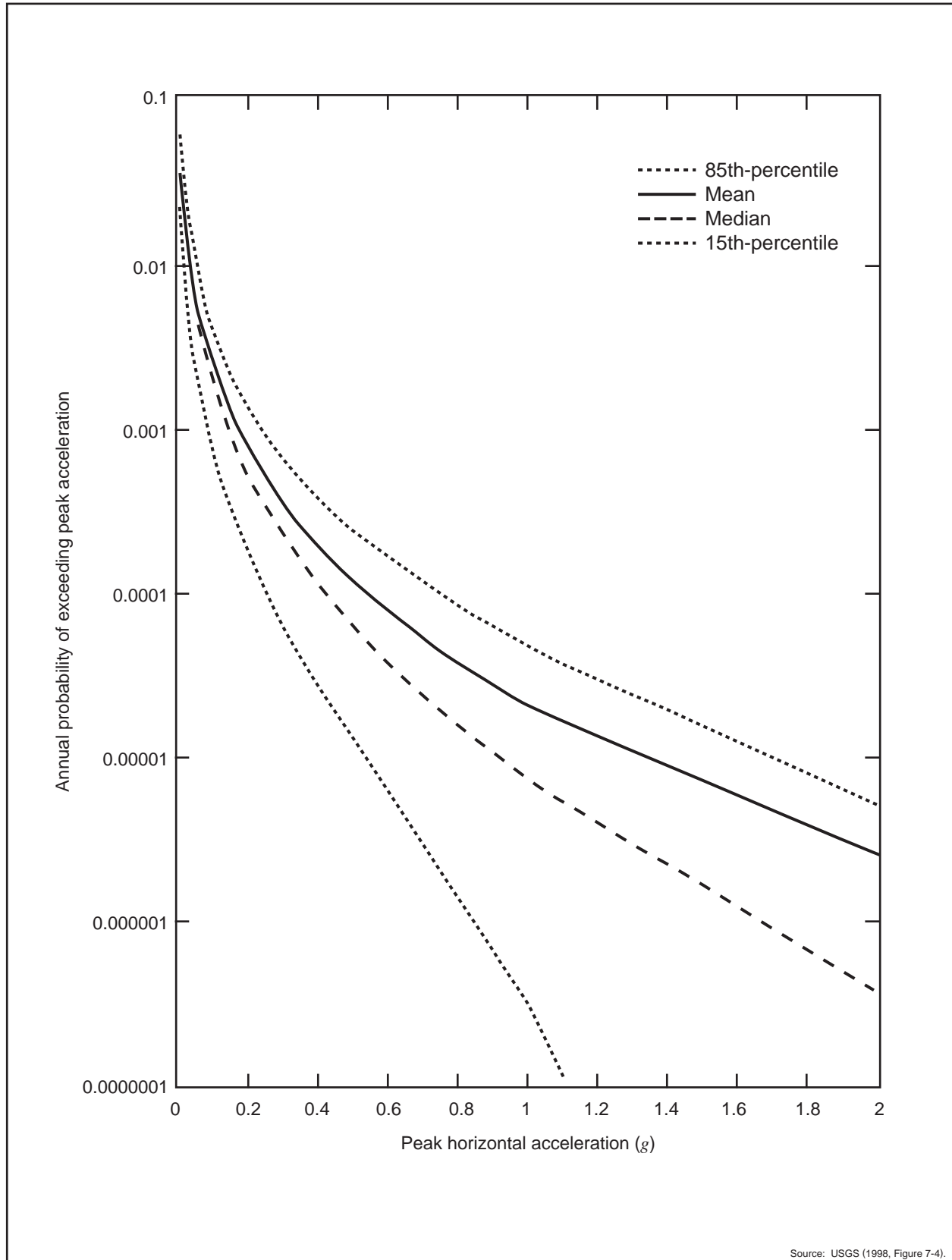
**13. Sandstorm.** Severe sandstorms could cause transporter derailments and sand buildup on structures. However, such events would be unlikely to initiate accidents with the potential for radiological release. Ma et al. (1992, page 3-11) reached a similar conclusion. Furthermore, it is assumed that DOE probably would curtail operations if local forecasts indicated the expected onset of high winds with potential to generate sandstorms (Kappes 1998, page 32). For these reasons, the analysis eliminated this event from further consideration.

**14. Seismic Activity, Earthquake (including subsidence, surface faults, uplift, subsurface fault, and static fracture).** DOE has selected the beyond-design-basis earthquake for detailed analysis. The seismic design basis for the repository specifies that structures (including the Waste Handling Building), systems, and components important to safety should be able to withstand the horizontal motion from an earthquake with a return frequency of once in 10,000 years (annual probability of occurrence of 0.0001) (Kappes 1998, page VII-3). A recent comprehensive evaluation of the seismic hazards associated with the site of the proposed repository (USGS 1998, all) concluded that a 0.0001-per-year earthquake would produce peak horizontal accelerations at the site of about 0.53g (mean value). Structures, systems, and components are typically designed with large margins over the seismic design basis to account for uncertainties in material properties, energy absorption, damping, and other factors. For nuclear powerplant structures, the methods for seismic design provide a factor of safety of 2.5 to 6 (Kennedy and Ravindra 1984, page R-53). In the absence of detailed design information, the analysis conservatively assumed that the Waste Handling Building would collapse at an acceleration level twice that associated with the design-basis earthquake, or 1.1g. Figure H-1 shows that this acceleration level would be likely to occur with a frequency of about  $2 \times 10^{-5}$  per year (mean value).

The Waste Treatment Building is not considered a safety-related structure. Accordingly, the seismic design basis for this building is to withstand an earthquake event with a return frequency of 1,000 years (annual exceedance probability of  $1 \times 10^{-3}$  per year) (TRW 1999b, page 14). Consistent with the assumption for the Waste Handling Building, it is assumed that the Waste Treatment Building would collapse during an earthquake that produced twice the design level acceleration. From Figure H-1, the design-basis acceleration for a  $1 \times 10^{-3}$  per year event is 0.18g. Thus, the building collapse is assumed to occur at an acceleration level of 0.36, which has an estimated return frequency of about  $2 \times 10^{-4}$  per year. The analysis retains these events as accident initiators, and evaluates the consequences in subsequent sections. The effects of other seismic-related phenomena included under this event (subsidence, surface faults, uplift, etc.) would be unlikely to produce greater consequences than those associated with the acceleration produced by the seismic event selected for analysis (complete collapse of the Waste Handling and Waste Treatment Buildings).

**15. Tornado.** The probability of a tornado striking the repository is estimated to be  $3 \times 10^{-7}$  (one chance in 10 million) based on an assessment of tornado strike probability for any point on the Nevada Test Site (DOE 1996d, page 4-146), which is adjacent to the proposed repository. This is slightly above the credibility level of  $1 \times 10^{-7}$  for accidents, as defined by DOE (DOE 1993, page 28). However, most tornadoes in the western United States have relatively modest wind speeds. For example, the probability of a tornado with wind speeds greater than 100 miles per hour is 0.1 or less (Ramsdell and Andrews 1986, page 41). Thus, winds strong enough to damage the Waste Handling Building are considered to be not credible.

Tornadoes can generate missiles that could penetrate structures at the repository, but radioactive material would be protected either by shipping casks, the Waste Handling Building with thick concrete walls, or the waste package. Therefore, tornado-driven missiles would not be a great hazard.



**Figure H-1.** Integrated seismic hazard results: summary hazard curves for peak horizontal acceleration.

**16. Volcanism, Ash Fall.** The potential for volcanic activity at the proposed repository site has been studied extensively. A recent assessment (Geomatrix and TRW 1996, page 4-46) estimates that the mean annual frequency of a volcano event that would intersect the repository footprint would be  $1.5 \times 10^{-8}$  per year (with 5 percent and 95 percent bounds of  $5 \times 10^{-10}$  and  $5 \times 10^{-8}$  per year), which is below the frequency of a credible event (DOE 1993, page 28). This result is consistent with a previous study of volcano activity at the site (DOE 1998b, all). Impacts from a regional volcanic eruption would be more likely; such an event could produce ash fall on the repository, and would be similar to the sandstorm event discussed above. Ash fall could produce a very heavy loading on the roof of the Waste Handling Building. Studies have concluded, however, that the worst case event would be an ash fall of 3 centimeters (1.2 inches) and analyses to date indicate that repository structures would not be affected by a 3-centimeter ash fall (DOE 1998b, Volume 1, pages 2-9).

**17. Sabotage.** The analysis separately considered sabotage (not listed in Table H-2) as a potential initiating event. This event would be unlikely to contribute to impacts from the repository. The repository would not represent an attractive target to potential saboteurs due to its remote location and the low population density in the area. Furthermore, security measures DOE would use to protect the waste material from intrusion and sabotage (TRW 1999b, pages 58 to 60) would make such attempts unlikely to succeed. At all times the waste material would be either in robust shipping or disposal containers or inside the Waste Handling Building, which would have thick concrete walls. On the basis of these considerations, DOE concluded that sabotage events would be unlikely at the repository. DOE expects that both the likelihood and consequences of sabotage events would be greater during transportation of the material to the repository (DOE 1997d, page 14). Appendix J presents the impacts of sabotage events during transportation.

Based on the external event assessment, DOE concluded that the only external event with a credible potential to release radionuclides of concern would be a large seismic event. This conclusion is supported by previous studies that screened out all external event accident initiators except seismic events (Ma et al. 1992, page 3-11; Jackson et al. 1984, pages 12 and 13). DOE is continuing to evaluate a few external events (Kappes 1998, page 33), and will examine the results of these evaluations to confirm the Draft EIS conclusions. If revisions are necessary, they will be provided in the Final EIS.

#### **H.2.1.4 Source Terms for Repository Accident Scenarios**

Following the definition of the accident scenarios as provided in previous sections, the analysis then estimated a source term for each accident scenario retained for analysis. The source term specification needed to include several factors, including the quantity of radionuclides released, the elevation of the release, the chemical and physical forms of the released radionuclides, and the energy (if any) of the plume that would carry the radionuclides to the environment. These factors would be influenced by the state of the material involved in the accident and the extent and type of damage estimated for the accident sequence. The estimate of the source term also considered mitigation measures, either active (for example, filtration systems) or passive (for example, local deposition of radionuclides or containment), that would reduce the amount of radioactive material released to the environment.

The analysis developed the source term for each accident scenario retained for evaluation. These include the accident scenarios retained from the internal events as listed in Table H-1 and the seismic event retained from the external event evaluation. Because many of the internal event-initiated accidents would involve drops of commercial spent nuclear fuel, the analysis considered the source term for these accidents as a group. Accordingly, source terms were developed for the following accident scenarios: commercial spent nuclear fuel drops, transporter runaway and derailment, DOE spent nuclear fuel drop, seismic event, and low-level waste drum failure.

#### H.2.1.4.1 Commercial Spent Nuclear Fuel Drop Accident Scenario Source Term

Commercial spent nuclear fuel contains more than 100 radioactive isotopes (SNL 1987, Appendix A). Not all of these isotopes, however, would be important in terms of a potential to cause adverse health effects (radiotoxicity) if released, and many would have decayed by the time the material arrived at the repository. Based on the characteristics of the radioactivity associated with an isotope (including type and energy of radioactive emissions, amount produced during the fissioning process, half-life, physical and chemical form, and biological impact if inhaled or ingested by a human), particular isotopes could be meaningful contributors to health effects if released. To determine the important radionuclides for an accident scenario consequence analysis, DOE consulted several sources. The Nuclear Regulatory Commission has identified a minimum of eight radionuclides in commercial spent nuclear fuel that “must be analyzed for potential accident release” (NRC 1997, page 7-6). Repository accident scenario evaluations (SNL 1987, pages 5-3 and 5-4) identified 14 isotopes (five of which were also on the Nuclear Regulatory Commission list) that contribute to “99 percent of the total dose consequence.” A more recent analysis (DOE 1996a, pages 6 to 9) lists 24 radionuclides (10 of which were not included in either of the other two lists) that are important for consequence analysis (99.9-percent cumulative dose for at least one organ). The DOE analysis also included carbon-14. Appendix A contains a list of 53 radionuclides, which includes the important isotopes discussed above. DOE used this longer list in the development of the source term for the accident scenario analyses.

Commercial spent nuclear fuel includes two primary types—boiling-water reactor and pressurized-water reactor spent fuel. For these commercial fuels, the radionuclide inventory depends on burnup (power history of the fuel) and cooling time (time since removal from the reactor). The EIS accident scenario analysis used “typical” fuels for each type. These typical fuels are representative of the majority of the fuel DOE would receive at the repository (see Appendix A). Table H-3 lists the characteristics of typical commercial spent nuclear fuel types.

A recent sensitivity study examined the consequences from accident scenarios involving bounding fuel types and illustrated the adequacy of selecting typical fuel types for this accident scenario analysis. Table H-4 lists the radionuclide inventory selected for estimating the accident scenario consequences for the fuel types selected (typical boiling-water reactor and pressurized-water reactor).

Commercial spent nuclear fuel damaged in the accidents evaluated in this EIS could release radionuclides from three different sources. These sources, and a best estimate of the release potential, are as follows:

**Table H-3.** Typical commercial spent nuclear fuel characteristics.<sup>a</sup>

Fuel type <sup>b</sup>	Cooling time (years)	Burnup (GWd/MTHM) <sup>c</sup>
PWR typical	25.9	39.56
BWR typical	27.2	32.2

a. Source: Appendix A.

b. PWR = pressurized-water reactor; BWR = boiling-water reactor.

c. GWd/MTHM = gigawatt-days per metric ton of heavy metal.

**H.2.1.4.1.1 Crud.** During reactor operation, crud (corrosion material) builds up on the outside of the fuel rod cladding and becomes radioactive from neutron activation. Five years after discharge from the reactor (the minimum age of any commercial spent nuclear fuel for acceptance at the repository), the dominant radioactive constituent in the crud is cobalt-60, which accounts for 98 percent of the activity (Sandoval et al. 1991, page 15). Cobalt-60 concentration measurements have been made on several boiling-water and pressurized-water reactor fuel rods; the results indicate that the maximum activity density is 0.0000094 curie per square centimeter for pressurized-water reactors and 0.000477 curie per square centimeter for boiling-water reactors (Sandoval et al. 1991, pages 14 and 15). The maximum values are about twice the average value over the length of the fuel rod (Sandoval et al. 1991, page 14). Accordingly, the values used in these source term determinations were 0.00005 for pressurized-water

**Table H-4.** Inventory used for typical reactor fuel (curies per assembly).<sup>a,b</sup>

Isotope	Pressurized-water reactor	Boiling-water reactor
Hydrogen-3	$9.8 \times 10^1$	$3.4 \times 10^1$
Carbon-14	$6.4 \times 10^{-1}$	$3.0 \times 10^{-1}$
Chlorine-36	$5.4 \times 10^{-3}$	$2.2 \times 10^{-3}$
Cobalt-60 <sup>c</sup>	$1.4 \times 10^1$	$2.0 \times 10^1$
Nickel-59	1.3	$3.5 \times 10^{-1}$
Nickel-63	$1.8 \times 10^2$	$4.6 \times 10^1$
Selenium-79	$2.3 \times 10^{-1}$	$7.9 \times 10^{-2}$
Krypton-85	$9.3 \times 10^2$	$2.9 \times 10^2$
Strontium-90	$2.1 \times 10^4$	$7.1 \times 10^3$
Zirconium-93	1.2	$4.8 \times 10^{-1}$
Niobium-93m	$8.2 \times 10^{-1}$	$3.5 \times 10^{-1}$
Niobium-94	$5.8 \times 10^{-1}$	$1.9 \times 10^{-2}$
Technetium-99	7.1	2.5
Rhodium-102	$1.2 \times 10^{-3}$	$2.8 \times 10^{-4}$
Ruthenium-106	$4.8 \times 10^{-3}$	$6.7 \times 10^{-4}$
Palladium-107	$6.3 \times 10^{-2}$	$2.4 \times 10^{-2}$
Tin-126	$4.4 \times 10^{-1}$	$1.5 \times 10^{-1}$
Iodine-129	$1.8 \times 10^{-2}$	$6.3 \times 10^{-3}$
Cesium-134	$1.6 \times 10^1$	3.4
Cesium-135	$2.5 \times 10^{-1}$	$1.0 \times 10^{-1}$
Cesium-137	$3.1 \times 10^4$	$1.1 \times 10^4$
Samarium-151	$1.9 \times 10^2$	$6.6 \times 10^1$
Lead-210	$2.2 \times 10^{-7}$	$9.4 \times 10^{-8}$
Radium-226	$9.3 \times 10^{-7}$	$3.7 \times 10^{-7}$
Radium-228	$1.3 \times 10^{-10}$	$4.7 \times 10^{-11}$
Actinium-227	$7.8 \times 10^{-6}$	$3.1 \times 10^{-6}$
Thorium-229	$1.7 \times 10^{-7}$	$6.1 \times 10^{-8}$
Thorium-230	$1.5 \times 10^{-4}$	$5.8 \times 10^{-5}$
Thorium-232	$1.9 \times 10^{-10}$	$6.9 \times 10^{-11}$
Protactinium-231	$1.6 \times 10^{-5}$	$6.0 \times 10^{-6}$
Uranium-232	$1.9 \times 10^{-2}$	$5.5 \times 10^{-3}$
Uranium-233	$3.3 \times 10^{-5}$	$1.1 \times 10^{-5}$
Uranium-234	$6.6 \times 10^{-1}$	$2.4 \times 10^{-1}$
Uranium-235	$8.4 \times 10^{-3}$	$3.0 \times 10^{-3}$
Uranium-236	$1.4 \times 10^{-1}$	$4.8 \times 10^{-2}$
Uranium-238	$1.5 \times 10^{-1}$	$6.2 \times 10^{-2}$
Neptunium-237	$2.3 \times 10^{-1}$	$7.3 \times 10^{-2}$
Plutonium-238	$1.7 \times 10^3$	$5.5 \times 10^2$
Plutonium-239	$1.8 \times 10^2$	$6.3 \times 10^1$
Plutonium-240	$2.7 \times 10^2$	$9.5 \times 10^1$
Plutonium-241	$2.0 \times 10^4$	$7.5 \times 10^3$
Plutonium-242	$9.9 \times 10^{-1}$	$4.0 \times 10^{-1}$
Americium-241	$1.7 \times 10^3$	$6.8 \times 10^2$
Americium-242/242m	$1.1 \times 10^1$	4.6
Americium-243	$1.3 \times 10^1$	4.9
Curium-242	8.7	3.8
Curium-243	8.3	3.1
Curium-244	$7.0 \times 10^2$	$2.5 \times 10^2$
Curium-245	$1.8 \times 10^{-1}$	$6.3 \times 10^{-2}$
Curium-246	$3.8 \times 10^{-2}$	$1.3 \times 10^{-2}$
Curium-247	$1.3 \times 10^{-7}$	$4.3 \times 10^{-8}$
Curium-248	$3.9 \times 10^{-7}$	$1.2 \times 10^{-7}$
Californium-252	$3.1 \times 10^{-8}$	$6.0 \times 10^{-9}$

a. Source: Appendix A, except cobalt-60.

b. Inventory numbers have been rounded to two significant figures.

c. Cobalt-60 inventory in crud, as calculated in this appendix.

reactors and 0.00025 for boiling-water reactors. Using the fuel rod dimensions and the number of rods per fuel assembly from Appendix A, these concentrations produce the following total inventory of cobalt-60 for a pressurized-water reactor fuel assembly at discharge:

$$\begin{aligned} \text{Cobalt-60 inventory} &= \text{fuel rod surface area per assembly} \times \text{cobalt-60 concentration} \\ \text{(per assembly)} &= \text{fuel rod diameter} \times \pi \\ &\quad \times \text{fuel rod length} \times \text{number of fuel rods per assembly} \\ &\quad \times \text{cobalt 60 concentration} \end{aligned}$$

For pressurized-water reactor assemblies, the corresponding values are (from Appendix A):

$$\begin{aligned} \text{Pressurized-water} &= 0.95 \text{ centimeters} \times 3.14 \\ \text{reactor cobalt-60} &\quad \times 366 \text{ centimeters} \times 264 \text{ rods} \\ \text{inventory} &\quad \times 0.00005 \text{ curie per square centimeter} \\ \text{(per assembly)} &\equiv 14 \text{ curies per pressurized-water reactor fuel assembly} \\ &\quad \text{(at reactor discharge)} \end{aligned}$$

For boiling-water reactor assemblies, the corresponding values are (from Appendix A):

$$\begin{aligned} \text{Boiling-water reactor} &= 1.25 \text{ centimeters} \times 3.14 \\ \text{cobalt-60 inventory} &\quad \times 366 \text{ centimeters} \times 55 \text{ rods} \\ \text{(per assembly)} &\quad \times 0.00025 \text{ curie per square centimeter} \\ &\equiv 20 \text{ curies per boiling-water reactor fuel assembly} \\ &\quad \text{(at reactor discharge)} \end{aligned}$$

The analysis used these concentrations, decayed to appropriate levels (25.9 years for pressurized-water reactor fuel and 27.2 years for boiling-water reactor fuel, from Table H-3), to obtain the final cobalt-60 inventory used in the source term determination.

The amount of crud that would be released from the surface of the fuel rod cladding is uncertain because there are very few data for the accident conditions of interest, and the physical condition of the crud can be highly variable (Sandoval et al. 1991, page 18). Two sources (NRC 1997, Table 7-1; NRC 1998, Table 4-1) recommend a release fraction of 1.0 (100 percent of the cobalt-60) for accident conditions; therefore, the EIS analysis assumed this value.

Following their release from the cladding, some crud particles would be retained by deposition on the surrounding surfaces (the fuel assembly cladding, spacer grids and structural hardware). The estimated fraction of released particles deposited on these surfaces would be 0.9 (SNL 1987, page 5-27), resulting in an escape fraction of 0.1. In accidents involving casks or canisters, additional surfaces represented by these components would offer surfaces for further plateout.

The inhalation radiation dose from cobalt-60 (or any radioactive particle) depends on the amount of particulate material inhaled into and remaining in the lungs (called the respirable fraction). The analysis assumed that the respirable fraction would be 0.05 (based on Wilmot 1981, page B-3). Therefore, the analysis assumed that the total cobalt-60 respirable airborne release fraction would be 0.005 (the escape fraction of 0.1 multiplied by the respirable fraction of 0.05) for accident scenarios involving commercial spent nuclear fuel assemblies.

**H.2.1.4.1.2 Fuel Rod Gap.** The space between the fuel rod cladding and the fuel pellets (called the *gap*) contains fission products released from the fuel pellets during reactor operation. The only

potentially important radionuclides in the gap are the gases tritium (hydrogen-3) and krypton-85, and the volatile radionuclides strontium-90, cesium-134, cesium-137, ruthenium-106, and iodine-129 (NRC 1997, page 7-6). The Nuclear Regulatory Commission recommends fuel rod release fractions (the fraction of the total fuel rod inventory) of 0.3 for tritium and krypton-85, 0.000023 for the strontium and cesium components, 0.000015 for ruthenium-106, and 0.1 for iodine under accident conditions that rupture the cladding (NRC 1997, page 7-6). The release fraction for the gases (tritium and krypton), as expected, would be rather high because most of the gas would be in the fuel rod gap and under pressure inside the fuel rod. The analysis also considered the fraction of the rods damaged in a given accident scenario. SNL (1987, page 6-19 *et seq.*) assumed that the fraction of damaged fuel pins in each assembly involved in a collision or drop accident scenario would be 20 percent. Another assessment (Kappes 1998, page 18) assumed that any drop of the fuel rods in a fuel assembly or basket of assemblies would result in failure of 10 percent of the fuel rods, regardless of the drop distance. Because neither value seems to have a strong basis, the EIS analysis assumed the more conservative 20-percent figure. For the particulate species released from the gap, the analysis applied a retention factor of 0.9 (escape factor of 0.1) to account for local deposition of the particles on the fuel assembly structures, consistent with SNL (1987, page 5-27). SNL (1987, page 5-28) also applies a similar factor to account for retention on the failed shipping cask structures for accident scenarios involving cask failure. However, the EIS analysis judged that this factor does not have a strong basis, especially because the actual mode of cask failure is unknown. For accident scenarios that could rupture the cask, surfaces on the cask structure might not be in the path of the released material and, therefore, would not be a potential deposition site. Furthermore, particulate material, which would escape local deposition on the fuel assembly surfaces, probably would be less susceptible to deposition on surfaces it encountered subsequently. Therefore, the analysis assumed no retention factor for cask structures. The final consideration is the fraction of remaining airborne particulates that would be respirable. No specific reference could be found to the volatile materials in the gap. The analysis conservatively assumed, therefore, that the respirable fraction would be 1.0.

**H.2.1.4.1.3 Fuel Pellet.** During reactor operation, the fuel pellets undergo cracking from thermal and mechanical stresses. This produces a small amount of pellet particulate material that contains radionuclides. The analysis assumed that the radionuclides are distributed evenly in the fuel pellets so that the fractional release of the pellet particulates is equivalent to the same fractional release of the total inventory of the appropriate radionuclides in the fuel. If the fuel cladding failed during an accident, a fraction of these particulates would be small enough (diameter less than 10 micrometers) for release to the atmosphere and would be respirable (small enough to remain in the lungs if inhaled). Sandia National Laboratories estimates this fraction to be 0.000001 (SNL 1987, page 5-26) based on experiments performed at Oak Ridge National Laboratory. The EIS used this value to develop source terms for the accident scenarios considered. Additional particulates could be produced by pulverization due to mechanical stresses imposed on the fuel pellets from the accident conditions. This pulverization factor has been evaluated in SNL (1987, page 5-17) and applied in Kappes (1998, page I-3). Based on experimental results involving bare fuel pellets, the analysis determined that the fraction likely to be pulverized into respirable particles would be proportional to the drop height (which is directly proportional to energy input) and would be:

$$2.0 \times 10^{-7} \times \text{energy partition factor} \times \text{unimpeded drop height (centimeters)} \quad (\text{Kappes 1998, page I-3}).$$

The energy partition factor is the fraction of the impact energy that is available for pellet pulverization. A large fraction of the impact energy is expended in deforming the fuel assembly structures and rupturing the fuel rod cladding. It has been estimated (SNL 1987, page 5-25) that the energy partition factor is 0.2.

As indicated above, some of the dispersible pellet particulates released in the accident could deposit on surfaces in the vicinity of the damaged fuel. Consistent with the particulate material considered above, the estimated fraction that would not deposit locally and would remain airborne would be 0.1 based on

SNL (1987, page 5-26). Based on these considerations, the respirable airborne release fraction produced from pulverization of the fuel pellets would be:

$$\begin{aligned}
 \text{Respirable airborne release fraction} &= 2 \times 10^{-7} \times \text{drop height (centimeters)} \\
 &\quad \times \text{energy partition factor} \times \text{fraction not deposited} \\
 &\quad \times \text{fuel rod damage fraction} \\
 &= 2 \times 10^{-7} \times \text{drop height} \\
 &\quad \times 0.2 \times 0.1 \\
 &\quad \times 0.2 \\
 &= 8 \times 10^{-10} \times \text{drop height}
 \end{aligned}$$

This result is reasonably consistent with the value of  $8 \times 10^{-7}$  from SAIC (1998, page 3-9), which is characterized as a bounding value for the respirable airborne release fraction for accident scenarios that would impose mechanical stress on fuel pellets for a range of energy densities (drop heights). This value would correspond to a drop from 1,000 centimeters (10 meters or 33 feet) based on the formulation above.

**H.2.1.4.1.4 Conclusions.** Table H-5 summarizes the source term parameters for commercial spent nuclear fuel drop accident scenarios, as discussed above.

**Table H-5.** Source term parameters for commercial spent nuclear fuel drop accident scenarios.

Radionuclide <sup>a</sup>	Location	Damage fraction	Release fraction	Fraction not deposited	Respirable fraction	Respirable airborne release fraction
Co-60	Clad surface	1.0	1.0	0.1	0.05	0.005
H-3, Kr-85, C-14	Gap	0.2	0.3	1.0	1.0	0.06
I-129	Gap	0.2	0.1	1.0	1.0	0.02
Cs-137, Sr-90	Gap	0.2	$2.3 \times 10^{-5}$	0.1	1.0	$4.6 \times 10^{-7}$
Ru-106	Gap	0.2	$1.5 \times 10^{-5}$	0.1	1.0	$3.0 \times 10^{-7}$
All solids	Gap (existing fuel fines)	0.2	$1.0 \times 10^{-6}$	0.1	1.0	$2.0 \times 10^{-8}$
All solids	Pellet-pulverization	0.2	$4.0 \times 10^{-8} \times h^b$	0.1	1.0	$8.0 \times 10^{-10} \times h^b$

a. Abbreviations: Co = cobalt; H = hydrogen (H-3 = tritium); Kr = krypton; C = carbon; I = iodine; Cs = cesium; Sr = strontium; Ru = ruthenium.

b. h = drop height in centimeters.

#### H.2.1.4.2 Transporter Runaway and Derailment Accident Source Term

This accident, as noted in Section H.2.1.3, would involve the runaway and derailment of the waste package transporter. It assumes the ejection of the waste package from the transporter during the event; the waste package would be split open by impact on the access tunnel wall. The calculated maximum impact speed would be 18 meters per second (38 miles per hour) (DOE 1997b, page 98). This analysis assumed that the source term from the damage to the 21 pressurized-water reactor fuel assemblies in the waste package is equivalent to a drop height that would produce the same impact velocity (equivalent to the same energy input). The equivalent drop height was computed from basic equations for the motion of a body falling under the influence of gravity:

$$\begin{aligned}
 \text{velocity} &= \text{acceleration} \times \text{time} \\
 \text{and,} \\
 \text{distance} &= \frac{1}{2} \times \text{acceleration} \times \text{time squared}
 \end{aligned}$$

where: velocity = velocity of the impact (18 meters per second)  
time = time required for the fall  
acceleration = acceleration due to gravity (9.8 meters per second squared)

By substitution,  
distance =  $\frac{1}{2} \times \text{acceleration} \times (\text{velocity} \div \text{acceleration})^2$   
=  $(\text{velocity})^2 \div (\text{acceleration} \times 2)$   
=  $(18)^2 \div (9.8 \times 2)$   
= 16 meters

Thus, the calculation of the source term for this accident scenario assumed a drop height of 16 meters and used the parameters in Table H-5 for the various nuclide groups.

#### **H.2.1.4.3 DOE Spent Nuclear Fuel Drop Accident Source Term**

Appendix A lists the various types of DOE spent nuclear fuel and high-level radioactive waste that the Department would place in the proposed repository. A review of the inventory indicates that the spent nuclear fuel from the Hanford Site (N-Reactor fuel) represents a large percentage of DOE spent nuclear fuel. The N-Reactor fuel also has one of the highest radionuclide inventories of any of the DOE spent fuels. Although a canister of naval spent nuclear fuel would have a higher radionuclide inventory than a canister of N-Reactor fuel (Appendix A, Table A-18), the amount of radioactive material that would be released from a naval canister during this hypothetical accident scenario would be less than the amount released from an N-Reactor fuel canister due to the highly robust design of naval fuel (Appendix A, Section A.2.2.5.3) (USN 1996, all). Therefore, DOE selected N-Reactor spent nuclear fuel material as the bounding form to represent the source term for accidents that would involve DOE material. The analysis derived the source term for accidents involving a drop of N-Reactor fuel from DOE (1995, page 5-88), which lists the estimated source term for a drop of a cask containing 1,000 kilograms (2,200 pounds) of N-Reactor fuel from a height of 4.6 meters (15 feet). For the repository accident scenario involving N-Reactor fuel, a total of 4,800 kilograms (10,600 pounds) of fuel would be involved in a multi-canister overpack drop (Appendix A) from a height of 6.3 meters (21 feet), as noted above. The analysis adjusted the DOE (1995, page 5-88) source term upward by a factor of 4.8 to account for the increased amount of material involved (4,800 kilograms as opposed to 1,000 kilograms), and by a factor of 1.37 to account for the increased drop height (6.3/4.6) because the analysis assumed the source term would be proportional to the energy input, which is proportional to the drop height. These two factors were applied to the DOE (1995, page 5-88) source term and the result is listed in Table H-6. The behavior of N-Reactor fuel during an accident is uncertain (Kappes 1998, page 15) and the Final EIS analysis might utilize a revised source term estimate based on the results of further studies of this fuel. Furthermore, DOE has not developed the requirements for receipt of the fuel at the repository. These requirements could influence the source term, as could the corresponding requirements for processing the fuel prior to shipment.

#### **H.2.1.4.4 Seismic Accident Scenario Source Term**

**Waste Handling Building.** In this event, as noted in Section H.2.1.3, the Waste Handling Building could collapse from a beyond-design-basis earthquake. Bare fuel assemblies being transferred during the event would be likely to drop to the floor and concrete from the ceiling could fall on the fuel assemblies, causing damage that could result in radioactive release, which would discharge to the atmosphere through the damaged roof. In addition, other radioactive material stored or being handled in the Waste Handling Building could be vulnerable to damage. To estimate the source term, the analysis evaluated the extent of damage to the fuel rods and pellets for the assemblies being transferred and then examined the other material that could be vulnerable.

**Table H-6.** Source term used for N-Reactor Mark IV fuel drop accident scenario analysis (curies).<sup>a</sup>

Radionuclide	Total release	Radionuclide	Total release	Radionuclide	Total release
Tritium (H <sub>3</sub> )	1.7×10 <sup>-2</sup>	Tin-119m	1.7×10 <sup>-8</sup>	Europium-154	8.3×10 <sup>-2</sup>
Carbon-14	2.6×10 <sup>-4</sup>	Tin-121m	3.0×10 <sup>-5</sup>	Uranium-234	1.7×10 <sup>-4</sup>
Iron-55	1.3×10 <sup>-3</sup>	Tin-126	5.6×10 <sup>-5</sup>	Uranium-235	5.7×10 <sup>-6</sup>
Nickel-59	1.4×10 <sup>-5</sup>	Stibium-125 (antimony)	2.4×10 <sup>-2</sup>	Uranium-236	3.3×10 <sup>-5</sup>
Nickel-63	1.7×10 <sup>-3</sup>	Stibium-126	7.9×10 <sup>-6</sup>	Uranium-238	1.4×10 <sup>-4</sup>
Cobalt-60	5.4×10 <sup>-2</sup>	Stibium-126m	5.6×10 <sup>-5</sup>	Neptunium-237	2.6×10 <sup>-5</sup>
Selenium-79	2.9×10 <sup>-5</sup>	Tellurium-125m	6.7×10 <sup>-3</sup>	Plutonium-238	7.9×10 <sup>-2</sup>
Krypton-85	2.4×10 <sup>-2</sup>	Iodine-129	2.3×10 <sup>-6</sup>	Plutonium-239	7.3×10 <sup>-2</sup>
Strontium-90	3.6	Cesium-134	2.3×10 <sup>-2</sup>	Plutonium-240	5.9×10 <sup>-2</sup>
Yttrium-90	3.6	Cesium-135	2.6×10 <sup>-5</sup>	Plutonium-241	4.3
Niobium-93m	7.2×10 <sup>-5</sup>	Cesium-137	4.9	Plutonium-242	4.9×10 <sup>-5</sup>
Zirconium-93	1.3×10 <sup>-4</sup>	Cerium-144	8.9×10 <sup>-5</sup>	Americium-241	1.7×10 <sup>-1</sup>
Technetium-99	9.7×10 <sup>-4</sup>	Praseodymium-144	8.9×10 <sup>-5</sup>	Americium-242	3.9×10 <sup>-4</sup>
Ruthenium-106	8.0×10 <sup>-4</sup>	Praseodymium-144m	1.1×10 <sup>-6</sup>	Americium-242m	3.9×10 <sup>-4</sup>
Palladium-107	6.7×10 <sup>-6</sup>	Promethium-147	2.4×10 <sup>-1</sup>	Americium-243	5.4×10 <sup>-5</sup>
Silver-110m	1.3×10 <sup>-8</sup>	Samarium-151	4.6×10 <sup>-2</sup>	Curium-242	3.2×10 <sup>-4</sup>
Cadmium-113m	1.6×10 <sup>-3</sup>	Europium-152	4.9×10 <sup>-4</sup>	Curium-244	2.4×10 <sup>-2</sup>

a. Source: DOE (1995, page 5-88), with adjustments as noted above.

The ceiling of the transfer cell, which would consist of concrete 20 to 25 centimeters (8 to 10 inches) thick, would be about 15 meters (50 feet) high (TRW 1999b, Attachment IV, Figure 13). Typical pressurized-water reactor fuel assemblies weigh 660 kilograms (1,500 pounds) each (see Appendix A). The assemblies are about 21 centimeters (8.3 inches) wide by about 410 centimeters (160 inches) long, for an effective cross-sectional area (horizontal) of 1 square meter (11 square feet) (SNL 1987, page 5-2). The weight of a single fuel assembly is roughly equivalent to a 25-centimeter-thick concrete block with a 1-square-meter cross-section [about 750 kilograms (1,700 pounds) based on a density of 2.85 grams per cubic centimeter (180 pounds per cubic foot) (CRC 1997, page 15-28)]. Thus, as a first approximation, the analysis assumed that the concrete blocks falling from the ceiling onto the fuel assemblies would produce about the same energy as the fuel assemblies falling from the same height.

Some of the energy imparted to the fuel assemblies from the falling debris would be absorbed in deforming the fuel assembly structures and, thus, would not be available to pulverize the fuel pellets. As evaluated above for falling fuel assemblies, this energy absorption factor would result in an estimated 20 percent of the energy being imparted to the pellets and the rest absorbed by the structure (SNL 1987, page 5-25). Finally, as noted above, the analysis used a 0.1 release factor (0.9 retention) to represent the retention of the released fuel particles by deposition on the cladding and other fuel assembly structures (SNL 1987, page 5-27). In addition, it assumed that additional retention would be associated with the concrete and other rubble that would be on top, or in the vicinity, of the fuel assemblies. It assumed this release factor would be 0.1 (0.9 retention) consistent with that used by SNL (1987, page 5-28) for retention by deposition on the cask and canister materials that surround the fuel assemblies during accident scenarios. It also assumed a fuel pellet pulverization factor of  $8 \times 10^{-10} \times h$ , the same as that used for fuel assembly drop accident scenarios. Thus, the overall pellet respirable airborne release fraction for the fuel pellet particulates is:

$$\begin{aligned}
 \text{Respirable airborne release fraction} &= 8 \times 10^{-10} \times \text{drop height (centimeters)} \times \text{rubble retention} \\
 &= 8 \times 10^{-10} \times 1,500 \times 0.1 \\
 &= 1.2 \times 10^{-6}
 \end{aligned}$$

Other radioactive materials either stored or being handled in the Waste Handling Building could also be at risk. For material in casks and canisters and waste packages, the analysis assumed that the damage

potential from falling debris would not be great enough to cause a large radionuclide release. This is based on the fact that canisters and casks are quite robust and that, even if the containers were breached by the energy of the impact, there would be very little energy remaining to cause fuel pellet pulverization. There could be, however, bare fuel assemblies exposed in the dryers and in disposal containers awaiting lid attachment. An estimated 375 bare pressurized-water reactor fuel assemblies could be exposed to falling debris (Montague 1999, page 1). The location of this material would be as follows:

- Assembly transfer system dryers: 25 pressurized-water reactor assemblies
- Disposal canister handling system welding stations: 346 pressurized-water reactor assemblies
- Transfer operations: four pressurized-water reactor assemblies

Because the concrete roof heights over these areas would be roughly the same as the assembly transfer system area in the Waste Handling Building [15 meters (50 feet)] where the analysis assumed the four bare pressurized-water reactor assemblies would be involved, the analysis assumed the pellet pulverization contribution to the source term to be equivalent to that for the fuel assemblies being transferred. The overall source term, then, was determined by assuming 375 typical pressurized-water reactor assemblies with the release fractions listed in Table H-5.

Boiling-water reactor fuel assemblies could be exposed at these areas, but the analysis evaluated only pressurized-water reactor fuel assemblies because they would result in a slightly higher source term under equivalent accident conditions and would be more likely to be involved because they would comprise a larger amount of material (see Appendix A) to be received at the repository. Thus, the source term for the seismic event would be 375 typical pressurized-water reactor fuel assemblies (Table H-4) with release fractions based on Table H-5.

**Waste Treatment Building.** It is assumed that the radionuclide concentration for the dry compactable waste in the Waste Treatment Building would be similar to that for power reactors (McFeely 1998, page 2). This material would consist of paper, plastic, and cloth with a specific activity of 0.025 curie per cubic meter (0.7 millicurie per cubic foot) (McFeely 1998, page 2). This activity would consist primarily of cobalt isotopes (primarily cobalt-60) representing 67 percent of the total activity, and cesium, which would contribute 28 percent of the total (McFeely 1999, all).

The Waste Treatment Building would operate a single shift per day, and would continuously process waste such that no large accumulation would occur. Because Waste Handling Building operations would be likely to involve three shifts per day (TRW 1999b, Section 6.2), the analysis assumed that three shifts of solid waste would accumulate before the Waste Treatment Building began its single-shift operation. The generation rate of solid compactible waste would be about 1,500 cubic meters (53,000 cubic feet) per year (DOE 1997a, page 32) or about 0.17 cubic meter (5.8 cubic feet) per hour. Thus, three shifts (24 hours) of Waste Handling Building operation would produce about 4.0 cubic meters (140 cubic feet) of solid compactible waste. The total radionuclide inventory in this waste would be:

Cobalt-60	=	4.0 cubic meters × 0.025 curie per cubic meters × 0.67 (cobalt-60 fraction)
	≅	0.07 curie
Cesium-137	=	4.0 cubic meters × 0.025 curie per cubic meters × 0.28 (cesium-137 fractions)
	≅	0.03 curie

The respirable airborne release fraction for a fire involving combustible low-level waste has been conservatively estimated at 0.4 (Mueller et al. 1996, page D-21). Thus, the respirable airborne release source term for the fire accident scenario would be:

$$\begin{array}{lcl} \text{Cobalt-60} & = & 0.07 \text{ curie} \times 0.4 = 0.028 \text{ curie} \\ \text{Cesium-137} & = & 0.03 \text{ curie} \times 0.4 = 0.012 \text{ curie} \end{array}$$

The assumed release height for the accident scenario is 2 meters (6.6 feet). This is the minimum release height for the consequences analysis and represents a ground-level release.

#### **H.2.1.4.5 Low-Level Waste Drum Failure Source Term**

As indicated in Section H.2.1.2, the most meaningful accident scenarios involving exposure to workers would be those related to puncture or rupture of waste drums that contained low-level waste. Such events could occur during handling operations and probably would involve the puncture of a drum by a forklift, or the drop of the drum during stacking and loading operations.

Two types of waste drums would contain the processed waste. Concentrated liquid waste would be mixed with cement and poured into 0.21-cubic-meter (55-gallon) drums. Compacted and noncompacted solid waste would also be placed in the same drums, which would, in turn, be placed in 0.32-cubic-meter (85-gallon) drums with the space between the two drums grouted. The probability of a drum failure was analyzed for these two drum types.

Following a drum failure, some fraction of the radionuclides in the waste would be released and workers in the immediate vicinity could be exposed to the material. The amount released would depend on the radionuclide concentration in the low-level waste material, the fraction of low-level waste released from the drum on its failure, and the respirable airborne release fraction from the released waste.

For liquid waste, the concentration of radionuclides is expected to be (McFeely 1998, page 3):

$$\begin{array}{lcl} \text{Cobalt-60} & = & 0.001 \text{ curie per cubic meter} \\ \text{Cesium-137} & = & 0.0015 \text{ curie per cubic meter} \end{array}$$

As noted in Section H.2.1.2, the evaporator would concentrate the liquid waste down to 10 percent of the original generated so the concentration of radionuclides in the waste would be increased to:

$$\begin{array}{lcl} \text{Cobalt 60} & = & 0.01 \text{ curie per cubic meter} \\ \text{Cesium-137} & = & 0.015 \text{ curie per cubic meter} \end{array}$$

The grouting operation would dilute this concentration somewhat by adding cement, but this dilution has been ignored for conservatism.

The total activity in a 0.21-cubic meter (55-gallon) drum would become:

$$\begin{array}{lcl} \text{Cobalt-60} & = & 0.01 \text{ curie per cubic meter} \times 0.21 \text{ cubic meter} \\ & \cong & 0.0021 \text{ curie per drum} \\ \\ \text{Cesium-137} & = & 0.015 \text{ curie per cubic meter} \times 0.21 \text{ cubic meter} \\ & \cong & 0.0032 \text{ curie per drum} \end{array}$$

For dry compacted waste, the total inventory in a 0.21-cubic-meter (55-gallon) drum would be

$$\begin{aligned}\text{Cobalt-60} &= 0.21 \text{ cubic meter} \times 0.025 \text{ curie per cubic meter} \times 0.67 \text{ (cobalt-60 fraction)} \\ &\cong 0.0035 \text{ curie}\end{aligned}$$

$$\begin{aligned}\text{Cesium-137} &= 0.21 \text{ cubic meter} \times 0.025 \text{ curie per cubic meter} \times 0.28 \text{ (cesium-137 fraction)} \\ &\cong 0.0015 \text{ curie}\end{aligned}$$

The estimated amount of material released from drums containing solid waste is 25 percent of the contents based on Mueller et al. (1996, page 94). Values from Mueller et al. (1996, all) were used for the respirable airborne release fraction. For dry waste, the recommended respirable airborne release fraction is 0.001. For grouted liquid waste, this fraction is determined by the following equation:

$$\text{Respirable airborne release fraction} = A \times D \times G \times H$$

where:

$$\begin{aligned}A &= \text{constant } (2.0 \times 10^{-11}) \text{ (Mueller et al. 1996, page D-25)} \\ D &= \text{material density [3.14 grams per cubic centimeter (196 pounds per cubic foot)]} \\ &\quad \text{(McFeely 1998, all)} \\ G &= \text{gravitational acceleration [980 centimeters (32.2 feet) per second squared]} \\ H &= \text{height of fall of the drum in the accident scenario}\end{aligned}$$

The assumed height of the fall is 2 meters (6.6 feet), which would be the approximate maximum lift height when the drum was stacked on another drum or placed on a carrier for offsite transportation. This same formula applies to drum puncture accident scenarios (Mueller et al. 1996, page D-30), and the 2-meter drop event would be equivalent in damage potential to a forklift impact at about 4.5 meters per second (10 miles per hour). The respirable airborne release fraction for this case then becomes:

$$\begin{aligned}\text{Respirable airborne release fraction} &= 2.0 \times 10^{-11} \times 3.14 \times 980 \times 200 \\ &\cong 1.23 \times 10^{-5}\end{aligned}$$

Based on these results, the worker risk would be dominated by accidents involving drums that contained dry waste because both the frequency of the event [0.59 versus 0.46 (Section H.2.1.2)] and the release fraction [ $1 \times 10^{-3}$  versus  $1.23 \times 10^{-5}$  (derived above)] would be greater. The total amount of airborne respirable material release (source term) for the risk-dominant dry waste accident scenario would be:

$$\begin{aligned}\text{Cobalt-60} &= 0.0035 \text{ curie (total drum inventory)} \times 0.25 \text{ (fraction released)} \\ &\quad \times 0.001 \text{ (respirable airborne release fraction)} \\ &\cong 8.5 \times 10^{-7} \text{ curies}\end{aligned}$$

$$\begin{aligned}\text{Cesium-137} &= 0.0015 \text{ curie (total drum inventory)} \times 0.25 \text{ (fraction released)} \\ &\quad \times 0.001 \text{ (respirable airborne release fraction)} \\ &\cong 3.8 \times 10^{-7} \text{ curies}\end{aligned}$$

The analysis assumed that, following normal industrial practice, workers would not be in the area beneath suspended objects. Accordingly, the nearest worker was assumed to be 5 meters (16 feet) from the impact area. Therefore, the volume assumed for dispersion of the material prior to reaching the worker would be 125 cubic meters (4,400 cubic feet), which represents the immediate vicinity of the accident

location [a volume approximately 5 meters (16 feet) by 5 meters by 5 meters]. The breathing rate of the worker would be 0.00035 cubic meter (about 0.012 cubic foot) per second (ICRP 1975, page 346).

#### **H.2.1.5 Assessment of Accident Scenario Consequences**

Accident scenario consequences were calculated as individual doses (rem), collective doses (person-rem), and latent cancer fatalities. The receptors considered were (1) the maximally exposed offsite individual, defined as a hypothetical member of the public at the point on the proposed repository land withdrawal boundary who would receive the largest dose from the assumed accident scenario (a minimum distance of 11 kilometers (7 miles)), (2) the maximally exposed involved worker, the hypothetical worker who would be nearest the spent nuclear fuel or high-level radioactive waste when the accident occurred, (3) the noninvolved worker, the hypothetical worker near the accident but not involved in handling the material, assumed to be 100 meters (about 330 feet) from the accident, and (4) the members of the public who reside within about 80 kilometers (50 miles) of the proposed repository.

For radiation doses below about 20 rem and low dose rates (below 10 rem per hour), potential health effects would be those associated with a chronic exposure or an increase in the risk of fatal cancer (ICRP 1991, Chapter 3) (see the discussion in Appendix F, Section F.1). The International Committee on Radiation Protection has recommended the use of a conversion factor of 0.0005 fatal cancer per person-rem for the general population for low doses, and a value of 0.0004 fatal cancer per person-rem for workers for chronic exposures. The higher value for the general population accounts in part for the fact that the general population contains young people, who are more susceptible to the effects of radiation. These conversion factors were used in the EIS consequence analysis. The latent cancer fatality caused by radiation exposure could occur at any time during the remaining lifetime of the exposed individual. As dose increases above about 15 rem over a short period (acute exposures), observable physical effects can occur, including temporary male sterility (ICRP 1991, page 15). At even higher acute doses (above about 500 rem), death within a few weeks is probable (ICRP 1991, page 16).

DOE used the MACCS2 computer program (Rollstin, Chanin, and Jow 1990, all; Chanin and Young 1998, all) and the radionuclide source terms for the identified accident scenarios in Section H.2.1.4 to calculate consequences to receptors. This program, developed by the U.S. Nuclear Regulatory Commission and DOE, has been widely used to compute radiological impacts from accident scenarios involving releases of radionuclides from nuclear fuel and radioactive waste. DOE used this program for offsite members of the public, the maximally exposed offsite individual, and the noninvolved worker. The MACCS2 program calculates radiological doses based on a sampling of the distribution of weather conditions for a year of site-specific weather data. Meteorological data were compiled at the proposed repository site from 1993 through 1997. This analysis used the weather conditions for 1993. The selection of 1993 was based on a sensitivity analysis that showed that, on the average, the weather conditions for 1993 produced somewhat higher consequences than those for the other years for most receptors, although the variation from year to year was small.

For exposure to inhaled radioactive material, it was assumed (in accordance with U.S. Environmental Protection Agency guidance) that doses would accumulate in the body for a total of 50 years after the accident (Eckerman, Wollbarst, and Richardson 1988, page 7). For external exposure (from ground contamination and contaminated food consumption), the dose was assumed to accumulate for 70 years (DOE 1993, page 21).

The MACCS2 program provides doses to selected receptors for a contiguous spectrum of site-specific weather conditions. Two weather cases were selected for the EIS: (1) a median weather case (designated at 50 percent) that represents the weather conditions that would produce median consequences to the

receptors, and (2) a 95 percent weather case that provides higher consequences that would only be exceeded 5 percent of the time.

The MACCS2 program is not suitable for calculating doses to receptors near the release point of radioactive particles [within about 100 meters (330 feet)]. For such cases, the analysis calculated involved worker dose estimates using a breathing rate of 0.00035 cubic meter (0.012 cubic foot) per second (ICRP 1975, page 346). For involved worker dose calculations from accident scenarios in the cask transfer and handling area, the analysis assumed that the worker would be a minimum of 4.6 meters (15 feet) from the location of the cask impact with the floor during the accident (normal industrial practice would preclude workers from being in the immediate vicinity of areas where heavy objects could strike the floor during lifting operations). Because of the perceived hazard following a breached cask, the analysis assumed that the worker would immediately vacate the area after observing that the cask had ruptured. Accordingly, the analysis assumed that the worker would breathe air containing airborne radioactive material from the ruptured cask for 10 seconds.

For involved worker doses from the drum handling accident scenario, the analysis assumed that the worker (a forklift operator) would be 3 meters (10 feet) from the drum rupture location, and would breathe air containing radioactive material from the ruptured drum for 30 seconds.

The involved worker dose estimates used the same dose conversion factors as those used by the MACCS2 program for inhalation exposure.

The analysis assumed that the population around the repository would be that projected for the year 2000 (see Appendix G, Table G-44). The exposed population would consist of individuals living within about 80 kilometers (50 miles) of the repository, including pockets of people who would reside just beyond the 80-kilometer distance. The dose calculations included impacts from the consumption of food contaminated by the radionuclide releases. The contaminated food consumption analysis used site-specific data on food production and consumption for the region around the proposed site (TRW 1997b, all). For conservatism, the analysis assumed no mitigation measures, such as post-accident evacuation or interdiction of contaminated foodstuffs. However, DOE would take appropriate mitigation actions in the event of an actual release.

The results of the consequence analysis are listed in Tables H-7 (for 50-percent weather) and H-8 (for 95-percent weather). These tables list doses in rem for individual receptors and in person-rem (collective dose to all exposed persons) for the 80-kilometer (50-mile) population around the site. For selected receptors, as noted, the tables list estimated latent cancer fatalities predicted to occur over the lifetime of the exposed receptors as a result of the calculated doses using the conversion factors described in this section. These estimates do not consider the accident frequency. For comparison, in 1995 the lifetime incidence of fatal cancer from all causes for Nevada residents was 0.24 (CDC 1998, page 215). Thus, the estimated latent cancer fatalities for the individual receptors from accidents would be very small in comparison to the cancer incidence from other causes. For the 28,000 persons living within 80 kilometers of the site (see Appendix G), 6,720 ( $28,000 \times 0.24$ ) would be likely to die eventually of cancer. The accident of most concern for the 95-percent weather conditions (earthquake, Table H-8, number 14) would result only in an estimated 0.0072 latent cancer fatality for this same population.

## **H.2.2 NONRADIOLOGICAL ACCIDENT SCENARIOS**

A potential release of hazardous or toxic materials during postulated operational accident scenarios at the repository would be very unlikely. Because of the large quantities of radioactive material, radiological considerations would outweigh nonradiological concerns. The repository would not accept hazardous waste as defined by the Resource Conservation and Recovery Act (40 CFR Parts 260 to 299). Some

**Table H-7.** Radiological consequences of repository operations accidents for median (50th-percentile) meteorological conditions.

Accident scenario <sup>a,b,c</sup>	Frequency (per year) <sup>a</sup>	Maximally exposed offsite individual			Population		Noninvolved worker		Involved worker	
		Dose (rem)	LCFi <sup>d</sup>		Dose (person-rem)	LCFp <sup>d</sup>	Dose (rem)	LCFi	Dose (rem)	LCFi
1. 6.9-meter drop of shipping cask in CTHA-61 BWR assemblies-no filtration	4.5×10 <sup>-4</sup>	1.9×10 <sup>-3</sup>	1.0×10 <sup>-6</sup>		5.5×10 <sup>-2</sup>	2.7×10 <sup>-5</sup>	9.4×10 <sup>-1</sup>	3.8×10 <sup>-4</sup>	76	3.0×10 <sup>-2</sup>
2. 7.1-meter drop of shipping cask in CTHA-26 PWR assemblies-no filtration	6.1×10 <sup>-4</sup>	2.3×10 <sup>-3</sup>	1.2×10 <sup>-6</sup>		6.6×10 <sup>-2</sup>	3.3×10 <sup>-5</sup>	1.1	4.4×10 <sup>-4</sup>	90	3.6×10 <sup>-2</sup>
3. 4.1-meter drop of shipping cask in CTHA-61 BWR assemblies- no filtration	1.4×10 <sup>-3</sup>	1.3×10 <sup>-3</sup>	6.5×10 <sup>-7</sup>		3.9×10 <sup>-2</sup>	2.0×10 <sup>-5</sup>	5.7×10 <sup>-1</sup>	2.3×10 <sup>-4</sup>	46	1.8×10 <sup>-2</sup>
4. 4.1-meter drop of shipping cask in CTHA-26 PWR assemblies-no filtration	1.9×10 <sup>-3</sup>	1.4×10 <sup>-3</sup>	7.0×10 <sup>-7</sup>		4.6×10 <sup>-2</sup>	2.3×10 <sup>-5</sup>	6.6×10 <sup>-1</sup>	2.6×10 <sup>-4</sup>	53	2.1×10 <sup>-2</sup>
5. 6.3-meter drop of MCO in CTS-10 N-Reactor fuel canisters-filtration	4.5×10 <sup>-4</sup>	3.7×10 <sup>-7</sup>	1.9×10 <sup>-10</sup>		1.1×10 <sup>-5</sup>	5.3×10 <sup>-9</sup>	1.1×10 <sup>-4</sup>	4.4×10 <sup>-8</sup>	(e)	(e)
6. 6.3-meter drop of MCO in CTS-10 N-reactor fuel canisters-no filtration	2.2×10 <sup>-7</sup>	1.2×10 <sup>-3</sup>	6.0×10 <sup>-7</sup>		3.4×10 <sup>-2</sup>	1.7×10 <sup>-5</sup>	3.6×10 <sup>-1</sup>	1.4×10 <sup>-4</sup>	(e)	(e)
7. 5-meter drop of transfer basket in ATS-8 PWR assemblies-filtration	1.1×10 <sup>-2</sup>	6.6×10 <sup>-7</sup>	3.3×10 <sup>-10</sup>		4.0×10 <sup>-4</sup>	2.0×10 <sup>-7</sup>	1.7×10 <sup>-4</sup>	6.8×10 <sup>-8</sup>	(e)	(e)
8. 5-meter drop of transfer basket in ATS-8 PWR assemblies-no filtration	2.8×10 <sup>-7</sup>	5.6×10 <sup>-4</sup>	2.8×10 <sup>-7</sup>		1.7×10 <sup>-2</sup>	8.6×10 <sup>-6</sup>	1.6×10 <sup>-1</sup>	6.4×10 <sup>-5</sup>	(e)	(e)
9. 7.6-meter drop of transfer basket in ATS-16 BWR assemblies-filtration	7.4×10 <sup>-3</sup>	5.1×10 <sup>-7</sup>	2.6×10 <sup>-10</sup>		2.9×10 <sup>-4</sup>	1.5×10 <sup>-7</sup>	1.3×10 <sup>-4</sup>	5.2×10 <sup>-8</sup>	(e)	(e)
10. 7.6-meter drop of transfer basket in ATS-16 BWR fuel assemblies-no filtration	1.9×10 <sup>-7</sup>	6.1×10 <sup>-4</sup>	3.1×10 <sup>-7</sup>		1.6×10 <sup>-2</sup>	8.2×10 <sup>-6</sup>	1.8×10 <sup>-1</sup>	7.2×10 <sup>-5</sup>	(e)	(e)
11. 6-meter drop of disposal container in DCHS-21 PWR assemblies-filtration	1.8×10 <sup>-3</sup>	1.8×10 <sup>-6</sup>	9.0×10 <sup>-10</sup>		1.0×10 <sup>-3</sup>	5.2×10 <sup>-7</sup>	5.0×10 <sup>-4</sup>	2.0×10 <sup>-7</sup>	(e)	(e)
12. 6-meter drop of disposal container in DCHS-21 PWR fuel assemblies-no filtration	8.6×10 <sup>-7</sup>	1.7×10 <sup>-3</sup>	8.5×10 <sup>-7</sup>		5.1×10 <sup>-2</sup>	2.5×10 <sup>-5</sup>	5.1×10 <sup>-1</sup>	2.0×10 <sup>-4</sup>	(e)	(e)
13. Transporter runaway and derailment in access tunnel-21 PWR assemblies-filtration-16-meter drop height equivalent	1.2×10 <sup>-7</sup>	4.3×10 <sup>-3</sup>	2.2×10 <sup>-6</sup>		1.1×10 <sup>-1</sup>	5.4×10 <sup>-5</sup>	1.5	6.0×10 <sup>-4</sup>	(f)	(f)
14. Earthquake - 375 PWR assemblies	2.0×10 <sup>-5</sup>	9.1×10 <sup>-3</sup>	4.6×10 <sup>-6</sup>		3.6×10 <sup>-1</sup>	1.8×10 <sup>-4</sup>	8.3	3.3×10 <sup>-3</sup>	(f)	(f)
15. Earthquake w/fire in WTB	2.0×10 <sup>-5</sup>	1.8×10 <sup>-5</sup>	9.0×10 <sup>-9</sup>		6.3×10 <sup>-4</sup>	3.2×10 <sup>-7</sup>	5.2×10 <sup>-3</sup>	2.1×10 <sup>-6</sup>	(f)	(f)
16. LLW drum rupture in WTB	0.59	6.1×10 <sup>-10</sup>	3.1×10 <sup>-13</sup>		2.1×10 <sup>-8</sup>	1.1×10 <sup>-11</sup>	1.4×10 <sup>-7</sup>	5.6×10 <sup>-11</sup>	7.0×10 <sup>-5</sup>	2.8×10 <sup>-8</sup>

- a. Source: Kappes (1998, all). These frequency estimates are highly uncertain due to the preliminary nature of the repository design and are provided only to show potential accident sequence credibility. They represent conservative estimates based on the approach taken in Kappes (1998, all).
- b. CTHA = Cask Transfer/Handling Area, CTS = Canister Transfer System, ATS = Assembly Transfer System, DCHS = Disposal Container Handling System, WTB = Waste Treatment Building.
- c. To convert meters to feet, multiply by 3.2808.
- d. LCFi is the likelihood of a latent cancer fatality for an individual who receives the calculated dose. LCFp is the number of cancers probable in the exposed population from the collective population dose (person-rem). These values were computed based on a conversion of dose in rem to latent cancers as recommended by the International Council on Radiation Protection as discussed in this section.
- e. For these cases, the involved workers are not expected to be vulnerable to exposure during an accident because operations are done remotely. Thus, involved worker impacts were not evaluated.
- f. For these events, involved workers would likely be severely injured or killed by the event; thus, no radiological impacts were evaluated. For the seismic event, as many as 39 people could be injured or killed in the Waste Handling Building, and as many as 36 in the Waste Treatment Building based on current staffing projections (TRW 1998c, pages 17 and 18).

**Table H-8.** Radiological consequences of repository operations accidents for unfavorable (95th-percentile) meteorological conditions.

Accident scenario <sup>a,b,c</sup>	Frequency (per year) <sup>d</sup>	Maximally exposed offsite individual		Population		Noninvolved worker		Involved worker	
		Dose (rem)	LCFi <sup>d</sup>	Dose (person-rem)	LCFp <sup>d</sup>	Dose (rem)	LCFi	Dose (rem)	LCFi
1. 6.9-meter drop of shipping cask in CTHA-61 BWR assemblies-no filtration	$4.5 \times 10^{-4}$	$7.2 \times 10^{-3}$	$3.5 \times 10^{-6}$	1.7	$8.6 \times 10^{-4}$	5.1	$2.0 \times 10^{-3}$	76	$3.0 \times 10^{-2}$
2. 7.1-meter drop of shipping cask in CTHA-26 PWR assemblies-no filtration	$6.1 \times 10^{-4}$	$8.0 \times 10^{-3}$	$4.0 \times 10^{-6}$	2.1	$1.1 \times 10^{-3}$	5.9	$2.4 \times 10^{-3}$	90	$3.6 \times 10^{-2}$
3. 4.1-meter drop of shipping cask in CTHA-61 BWR assemblies-no filtration	$1.4 \times 10^{-3}$	$4.3 \times 10^{-3}$	$2.2 \times 10^{-6}$	1.3	$6.5 \times 10^{-4}$	3.1	$1.2 \times 10^{-3}$	46	$1.8 \times 10^{-2}$
4. 4.1-meter drop of shipping cask in CTHA-26 PWR assemblies-no filtration	$1.9 \times 10^{-3}$	$5.2 \times 10^{-3}$	$2.6 \times 10^{-6}$	1.5	$7.8 \times 10^{-4}$	3.5	$1.4 \times 10^{-3}$	53	$2.1 \times 10^{-2}$
5. 6.3-meter drop of MCO in CTS-10 N-Reactor fuel canisters-filtration	$4.5 \times 10^{-4}$	$1.2 \times 10^{-6}$	$6.0 \times 10^{-10}$	$2.6 \times 10^{-4}$	$1.3 \times 10^{-7}$	$3.3 \times 10^{-4}$	$1.3 \times 10^{-7}$	(e)	(e)
6. 6.3-meter drop of MCO in CTS-10 N-reactor fuel canisters-no filtration	$2.2 \times 10^{-7}$	$4.3 \times 10^{-3}$	$2.2 \times 10^{-6}$	$8.6 \times 10^{-1}$	$4.3 \times 10^{-4}$	1.1	$4.4 \times 10^{-4}$	(e)	(e)
7. 5-meter drop of transfer basket in ATS-8 PWR assemblies- filtration	$1.1 \times 10^{-2}$	$2.5 \times 10^{-6}$	$1.3 \times 10^{-9}$	$3.3 \times 10^{-2}$	$1.6 \times 10^{-5}$	$4.6 \times 10^{-4}$	$1.8 \times 10^{-7}$	(e)	(e)
8. 5-meter drop of transfer basket in ATS-8 PWR assemblies-no filtration	$2.8 \times 10^{-7}$	$2.1 \times 10^{-3}$	$1.1 \times 10^{-6}$	$5.6 \times 10^{-1}$	$2.8 \times 10^{-4}$	$4.6 \times 10^{-1}$	$1.8 \times 10^{-4}$	(e)	(e)
9. 7.6-meter drop of transfer basket in ATS-16 BWR assemblies-filtration	$7.4 \times 10^{-3}$	$2.1 \times 10^{-6}$	$1.1 \times 10^{-9}$	$2.4 \times 10^{-2}$	$1.2 \times 10^{-5}$	$3.8 \times 10^{-4}$	$1.5 \times 10^{-7}$	(e)	(e)
10. 7.6-meter drop of transfer basket in ATS-16 BWR fuel assemblies-no filtration	$1.9 \times 10^{-7}$	$2.2 \times 10^{-3}$	$1.1 \times 10^{-6}$	$5.1 \times 10^{-1}$	$2.6 \times 10^{-4}$	$5.1 \times 10^{-1}$	$2.0 \times 10^{-4}$	(e)	(e)
11. 6-meter drop of disposal container in DCHS-21 PWR assemblies-filtration	$1.8 \times 10^{-3}$	$7.3 \times 10^{-6}$	$3.7 \times 10^{-9}$	$8.6 \times 10^{-2}$	$4.3 \times 10^{-5}$	$1.3 \times 10^{-3}$	$5.2 \times 10^{-7}$	(e)	(e)
12. 6-meter drop of disposal container in DCHS-21 PWR fuel assemblies-no filtration	$8.6 \times 10^{-7}$	$6.1 \times 10^{-3}$	$3.1 \times 10^{-6}$	1.6	$8.0 \times 10^{-4}$	1.3	$5.2 \times 10^{-4}$	(e)	(e)
13. Transporter runaway and derailment in access tunnel-21 PWR assemblies-filtration-16-meter drop height equivalent	$1.2 \times 10^{-7}$	$1.3 \times 10^{-2}$	$6.5 \times 10^{-6}$	3.2	$1.6 \times 10^{-3}$	3.9	$1.6 \times 10^{-3}$	(f)	(f)
14. Earthquake - 375 PWR assemblies	$2.0 \times 10^{-5}$	$3.2 \times 10^{-2}$	$1.6 \times 10^{-5}$	14	$7.2 \times 10^{-3}$	7.0	$2.8 \times 10^{-2}$	(f)	(f)
15. Earthquake w/fire in WTB	$2.0 \times 10^{-4}$	$5.8 \times 10^{-5}$	$2.9 \times 10^{-8}$	2.1	$1.1 \times 10^{-5}$	$5.2 \times 10^{-3}$	$2.1 \times 10^{-6}$	(f)	(f)
16. LLW drum rupture in WTB	0.59	$1.9 \times 10^{-9}$	$9.5 \times 10^{-13}$	$7.5 \times 10^{-7}$	$3.7 \times 10^{-10}$	$1.4 \times 10^{-7}$	$5.6 \times 10^{-11}$	$7.0 \times 10^{-5}$	$2.8 \times 10^{-8}$

- a. Source: Kappes (1998, all). These frequency estimates are highly uncertain due to the preliminary nature of the repository design and are provided only to show potential accident sequence credibility. They represent conservative estimates based on the approach taken in Kappes (1998, all).
- b. CTHA = Cask Transfer/Handling Area, CTS = Canister Transfer System, ATS = Assembly Transfer System, DCHS = Disposal Container Handling System, WTB = Waste Treatment Building.
- c. To convert meters to feet, multiply by 3.2808.
- d. LCFi is the likelihood of a latent cancer fatality for an individual who receives the calculated dose. LCFp is the number of cancers probable in the exposed population from the collective population dose (person-rem). These values were computed based on a conversion of dose in rem to latent cancers as recommended by the International Council on Radiation Protection, as discussed in this section.
- e. For these cases, the involved workers are not expected to be vulnerable to exposure during an accident since operations are done remotely. Thus, involved worker impacts were not evaluated.
- f. For these events, involved workers would likely be severely injured or killed by the event; thus, no radiological impacts were evaluated. For the seismic event, as many as 39 people could be injured or killed in the Waste Handling Building, and as many as 36 in the Waste Treatment Building based on current staffing projections (TRW 1998c, pages 17 and 18).

potentially hazardous metals such as arsenic or mercury could be present in the high-level radioactive waste. However, they would be in a solid glass matrix that would make the exposure of workers or members of the public from operational accidents highly unlikely. Appendix A contains more information on the inventory of potentially hazardous materials.

Some potentially nonradioactive hazardous or toxic substances would be present in limited quantities at the repository as part of operational requirements. Such substances would include liquid chemicals such as cleaning solvents, sodium hydroxide, sulfuric acid, and various solid chemicals. These substances are in common use at other DOE sites. Potential impacts to workers from normal industrial hazards in the workplace including workplace accidents were derived from DOE accident experience at other sites. These impacts include those from accident scenarios involving the handling of hazardous materials and toxic substances as part of typical DOE operations. Thus, the industrial health and safety impacts to workers include impacts to workers from accidents involving such substances.

Impacts to members of the public would be unlikely because the hazardous materials would be mostly liquid and solid so that a release would be confined locally. (For example, chlorine used at the site for water treatment would be in powder form, so a gaseous release of chlorine would be unlikely. Furthermore, the repository would not use propane as a heating fuel, so no potential exists for propane explosions or fires.) The potential for hazardous chemicals to reach surface water during the Proposed Action would be limited to spills or leaks followed immediately by a rare precipitation or snow melt event large enough to generate runoff. Throughout the project, DOE would install engineered measures to minimize the potential for spills or releases of hazardous chemicals and would comply with written plans and procedures to ensure that, if a spill did occur, it would be properly managed and remediated. The Spill Prevention Control and Countermeasures Plan that would be in place for Yucca Mountain activities is an example of the plans DOE would follow under the Proposed Action.

The construction phase could generate as many as 3,500 drums [about 730 cubic meters (26,000 cubic feet)] of solid hazardous waste, and emplacement operations could generate as much as 100 cubic meters (3,500 cubic feet) per year (TRW 1999b, Section 6.1). Maintenance operations and closure would generate similar or smaller waste volumes. DOE would accumulate this waste in onsite staging areas in accordance with the regulations of the Resource Conservation and Recovery Act. Emplacement and maintenance operation could generate as many as 2,700 liters (1,700 gallons) of liquid hazardous waste annually (TRW 1999b, Section 6.1). The construction and closure phases would not generate liquid hazardous waste. The generation, storage, packaging, and shipment off the site of solid and liquid hazardous waste would present a very small potential for accidental releases and exposures of workers. Although a specific accident scenario analysis was not performed for these activities, the analysis of human health and safety (see Chapter 4, Section 4.1.7.3) included these impacts to workers implicitly through the use of a data base that includes impacts from accidents involving hazardous and toxic materials. Impacts to members of the public would be unlikely.

### **H.3 Accident Scenarios During Retrieval**

During retrieval operations, activities at the repository would be essentially the reverse of waste package emplacement, except operations in the Waste Handling Building would not be necessary because the waste packages would not be opened. The waste packages would be retrieved remotely from the emplacement drifts, transported to the surface, and transferred to a Waste Retrieval Storage Facility (TRW 1999b, Attachment I). This facility would include a Waste Retrieval Transfer Building where the waste packages would be unloaded from the transporter, transferred to a concrete storage unit, and moved to a concrete storage pad. The storage pad would be a 24- by 24-meter (80- by 80-foot) pad, about 1 meter (3.3 feet) thick, which probably would be about 3 kilometers (2 miles) over flat terrain from the

North Portal. Each storage pad would contain 14 waste packages. The number of pads required would depend on how many waste packages would be retrieved.

Because retrieval operations would be essentially the reverse of emplacement operations, accidents involving the disposal container during emplacement bound the retrieval operation. The bounding accident scenario during emplacement of the disposal container would be transporter runaway and derailment in the access tunnel (see Section H.2.1.4). This accident scenario would also bound accident scenarios during retrieval.

During storage, no credible accidents resulting in radioactive release of any measurable consequence would be expected to occur. This prediction is based on an evaluation of above-ground dry storage accident scenarios at the commercial sites under similar conditions, as evaluated in Appendix K.

In view of these considerations, DOE has concluded that the waste transporter derailment and the rockfall accident scenarios analyzed in Section H.2 would bound accident impacts during retrieval.

## **H.4 Accident Scenarios During Monitoring and Closure**

During monitoring and closure activities, DOE would not move the waste packages, with the possible exception of removing a container from an emplacement drift for examination or drift maintenance. Such operations could result in a transporter runaway and derailment accident, but the frequency of release from such an event would be extremely low, as would the consequences, resulting in minimal risk. Thus, DOE expects the radiological impacts from operations during monitoring and closure to be very small.

## **H.5 Accident Scenarios for Inventory Modules 1 and 2**

Inventory Modules 1 and 2 are alternative inventory options that the EIS considers. These modules involve the consideration of additional waste material for emplacement in the repository. They would involve the same waste and handling activities as those for the Proposed Action, but the quantity of materials received would increase, as would the period of emplacement operations. The analysis assumed the receipt and emplacement rates would remain the same as those for the Proposed Action. Therefore, DOE expects the accident impacts evaluated for the Proposed Action to bound those that could occur for Inventory Modules 1 and 2 because the same set of operations would be involved.

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